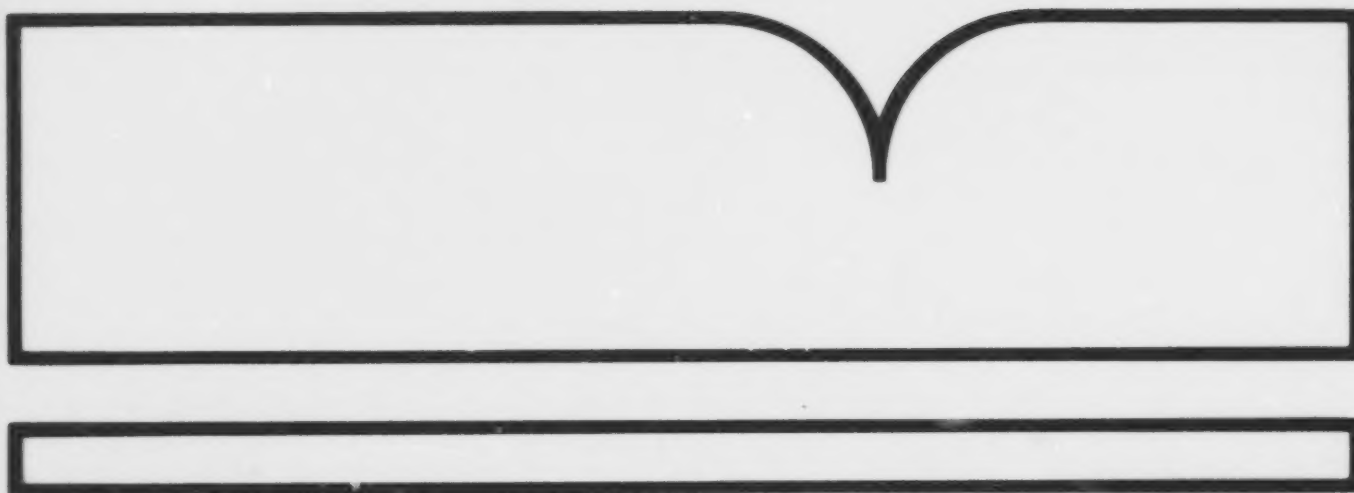


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Demonstrating Compliance with 40 CFR Part 61, Subpart 1

(U.S.) Office of Radiation Programs, Washington, DC

Oct 89



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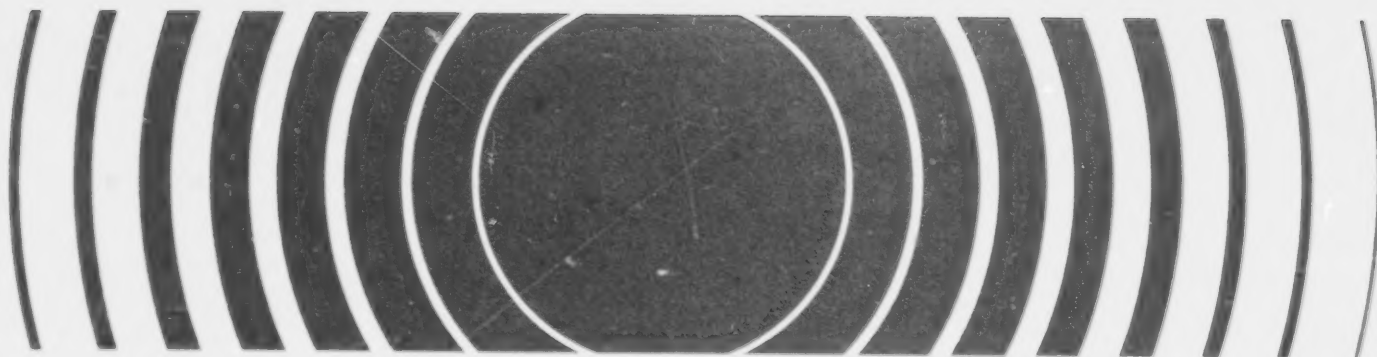
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National Emission Standards
for Hazardous Air Pollutants

Background Information Document:

Procedures Approved for
Demonstrating Compliance
with 40 CFR Part 61, Subpart I

Office of Radiation Programs
U.S. Environmental Protection Agency
Washington, DC

October 1989

*(National
Emission
Standards
for Hazardous
Air Pollutants)*

PREFACE

The purpose of this document is to provide information supporting the procedures developed for NRC-licensed and non-DOE Federal facilities to use in demonstrating compliance with the radionuclides NESHAP (40 CFR 61, Subpart I). Specifically, this Background Information Document has the following objectives:

- provide the basis for the calculational and analytical methods approved for determining emissions; and
- provide the basis for the procedures approved for demonstrating compliance with the dose limits of the standard.

This document comprises four chapters. Chapter 1 provides background information on the history of the rulemaking and summarizes the major provisions of the NESHAP. Chapter 2 describes the types of facilities covered by the NESHAP. Since the potential for releasing radioactive materials to the atmosphere depends on the specific radionuclides used, the processing and handling that they undergo, and the effluent controls which are used, the information presented focuses on these factors. Chapter 3 presents the basis for the methods approved to determine emissions. Particular emphasis is given to the derivation and use of emission factors* to estimate the quantities of material handled that

*As used in this document, an emission factor is defined as the pre-effluent control fraction of a radioactive material that becomes airborne. Thus, the quantity of material that is released to the atmosphere is the product of the quantity that becomes airborne and the appropriate effluent control adjustment factor.

become airborne and to the adjustment factors for effluent controls, which may be applied to these quantities to estimate emissions. Within the restrictions given, these emission factors and effluent control adjustment factors may be used in lieu of measured release rates in determining compliance with the standard. Chapter 4 presents the basis for the procedures approved for determining compliance. References are presented at the end of each chapter. Additional references are listed in Appendix A.

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CHAPTER 1. INTRODUCTION

1.1 BACKGROUND

On December 27, 1979, the Administrator of the Environmental Protection Agency (EPA) listed radionuclides as hazardous air pollutants subject to regulation under Section 112 of the Clean Air Act (FR79). Three National Emission Standards for Hazardous Air Pollutants (NESHAPS) were promulgated on February 6, 1985, regulating radionuclide emissions from Department of Energy (DOE) facilities, Nuclear Regulatory Commission- (NRC-) licensed and non-DOE Federal facilities, and elemental phosphorus plants (FR85a). Two additional radionuclide NESHAPS, covering radon-222 emissions from underground uranium mines and licensed uranium mill tailings, were promulgated on April 17, 1985, and September 24, 1986, respectively (FR85b, FR86).

The EPA's basis for the radionuclide NESHAPS (including its decision not to propose NESHAPS for certain categories of facilities) was challenged in lawsuits filed by the Sierra Club and the Natural Resources Defense Council (NRDC). While these suits were under adjudication, the U.S. Court of Appeals for the District of Columbia issued a decision finding that the EPA's NESHAP for vinyl chloride was defective in that costs had been improperly considered in setting the standard. Following the Court's order to review the potential effect of the vinyl chloride decision on other standards, the EPA determined that costs had been considered in many rulemakings on radionuclide emissions. On December 9, 1987, the Court accepted the EPA's proposal to leave the existing radionuclide NESHAPS in place while the Agency reconsider the standards. In the interim, the suits filed by the Sierra Club and the NRDC have been placed in abeyance. As a result of the reconsideration, the EPA has promulgated a NESHAP for NRC-licensed and non-DOE Federal Facilities.

This Background Information Document deals only with the compliance procedures for implementing the radionuclide NESHAP covering NRC-licensed and non-DOE Federal facilities (40 CFR Part 61, Subpart I). A separate Environmental Impact Statement presents the risks posed by these facilities and the Agency's rationale for the dose standard (EPA89).

This Background Information Document explains the basis for the procedures for demonstrating compliance with the dose limits. The compliance procedures include methods for determining the quantities of radionuclides released to the atmosphere and methods for estimating the resulting doses to the most exposed individual.

1.2 SUMMARY OF THE NESHAP

1.2.1 Applicability

The NESHAP applies to facilities licensed by the NRC or an Agreement State and all Federal facilities, except those owned or operated by the DOE, that use or possess unsealed radiation sources. The NESHAP does not apply to facilities regulated under 40 CFR Part 191 Subpart B, to low-energy accelerators, or to facilities that use or possess only sealed radiation sources.

The facilities covered by the NESHAP and the activities at these facilities involving the use of radioactive materials are quite diverse. They include both NRC material licensees and facilities engaged in the uranium fuel cycle. NRC material licensees include radiopharmaceutical suppliers and medical users, government, academic and private research facilities, test and research reactors, low-level waste shippers and disposal sites, and other suppliers and users of radionuclides. In total, the NESHAP

applies to an estimated 6,000 government, academic, medical, and industrial facilities.

1.2.2 The Standard

The NESHAP limits annual radionuclide emissions to the atmosphere from these facilities to such quantities that will not result in any member of the public receiving an effective dose equivalent in excess of 10 millirem per year (mrem/yr). Further, not more than 3 mrem/yr effective dose equivalent may be caused by isotopes of iodine. The standard specifically excludes doses caused by radon-220 or radon-222 and their decay products that are formed after release.

Facilities covered by the NESHAP are also subject to the reporting and approval requirements of 40 CFR Part 61, Subpart I, Sections 61.104(a) and 61.106(a) of Part 61. However, Sections 61.104(b) and 61.106(b) of Subpart I exempt from these requirements any facility that, using the specified procedures, demonstrates that its total emissions do not cause any member of the public to receive a dose greater than 10 percent of the limits of the standard. Further, the approval requirements are waived if, again using the specified procedures, the emissions from a newly constructed or modified facility will not cause any member of the public to receive a dose in excess of 1 percent of the standard.

1.2.3 Demonstrating Compliance

The standard limits doses to the most exposed member of the public. Dose is a complicated function of the quantity of each radionuclide emitted; the physical configuration of the facility releasing the material; the dispersion, transport, and build-up

of the radionuclides in the soil and foodstuffs; and the proximity to the facility of individuals and farms producing foodstuffs.

While, in principle, the doses resulting from the release of radionuclides to the atmosphere can be determined by environmental monitoring, valid measurements are often impossible to obtain. At the levels consistent with the limit of the standard, concentrations of many radionuclides will be below the minimum detection level of even state-of-the-art measurement technology. Even when the concentrations can be adequately measured, it may not be possible to distinguish the portion attributable to the emissions from that which is due to background radioactivity.

Therefore, compliance with the limit of the standard is to be demonstrated using EPA-approved methods of determining emissions and EPA-approved procedures for estimating the resulting doses. The NESHAP includes approved calculational and analytical methods for determining emissions, and a number of alternative procedures for determining doses. Other methods and procedures (including those based on environmental measurements) must be submitted to the EPA for approval before they are used.

REFERENCES

- EPA89 U.S. Environmental Protection Agency, Risk Assessments: Environmental Impact Statement for NESHAPS - Radionuclides, Volume 2, Office of Radiation Programs, Washington, DC, September 1989.
- FR79 44 Federal Register, 76738, December 27, 1979.
- FR85a 50 Federal Register, 5190-5200, February 6, 1985.
- FR85b 50 Federal Register, 15386-15394, April 17, 1985.
- FR86 51 Federal Register, 34056-34067, September 24, 1986.

CHAPTER 2. FACILITY DESCRIPTIONS

2.1 INTRODUCTION

The NESHAP applies to approximately 6,000 NRC-licensed and non-DOE Federal facilities that possess unsealed sources of radioactive materials. The NRC-licensed facilities include material licensees and facilities engaged in the uranium fuel cycle. NRC-licensed facilities include facilities licensed by the Agreement States but exclude low-energy accelerators and facilities regulated under 40 CFR Part 191, Subpart B.

The major types of facilities covered by the standard are described in the following sections. The discussion focuses on the physical forms of the radionuclides used and the handling and processing that the materials undergo. These factors are major determinants of the quantities of materials handled that become airborne.

2.2 NRC MATERIAL LICENSEES

2.2.1 Users and Producers of Radionuclides for Medical Purposes

The users and producers of radioactive materials for medical purposes constitute by far the largest category of facilities handling unsealed radioactive sources. Approximately two-thirds of the 6,000 facilities covered by the NESHAP are engaged in some aspect of the production and distribution of radiopharmaceuticals or in the medical application of these materials. Medical uses of radiopharmaceuticals include biomedical research and patient administration of radiopharmaceuticals for both diagnostic and therapeutic purposes.

2.2.1.1 Radiopharmaceutical Users

The types of facilities that use radionuclides for medical purposes include hospitals, clinics, and biomedical research facilities. The radionuclides used directly in patient therapy and diagnosis are termed "radiopharmaceuticals," while those used in research are referred to as "radionuclides." For simplicity, the term "radiopharmaceuticals" will be used to refer to the radioactive materials used in both patient administration and research.

The radiopharmaceuticals used at medical facilities occur in all three basic physical states: solid, liquid, and gas. The physical state of a particular radiopharmaceutical product is determined by (1) the chemical form of the radionuclide and (2) the solution or other mixture, if any, in which the radionuclide is dispensed. Both the radionuclide and the substance in which it is mixed are chosen to suit specific therapeutic, diagnostic, and research purposes.

The mixing of the radionuclide with some other substance means that the physical state of a radiopharmaceutical product may be different than the physical state of the radionuclide itself. In this document, discussions of the form of a particular radionuclide refer to the radionuclide product. The physical states of these products are important in assessing the potential for airborne release.

Most radionuclides used in medical facilities occur in liquid form. These liquids may be administered either orally or intravenously. Orally administered radionuclides are usually in the form of aqueous solutions. Many of these chemicals are ionic salts and thus occur in liquid form as saline solutions. Radionuclides that are administered intravenously may occur as solutions, colloids, or suspensions.

Solutions consist of molecules of solids or gaseous substances dissolved in a liquid. Colloids involve the dispersion of larger particles (on the order of 10 nanometers to 1 micrometer in diameter) in a liquid medium; the larger particles are prevented from aggregating and settling by being coated with a layer of gelatin (as is done with gold-198). Suspensions are similar to colloids but involve the radionuclide labeling of still larger particles (greater than 10 micrometers in diameter) of substances such as human serum albumin.

Gaseous radionuclides usually occur naturally in elemental form (e.g., xenon-133), and are administered to patients as a pure gas or as a gas diluted by air. Patients normally inhale the gas from a bag or from a gas "generator" through a respirator.

Solid radionuclides occur as gelatin capsules containing liquid solutions of the radionuclide chemical. In some cases, the solution is absorbed in dry filler material. Solid radionuclides are administered orally to patients.

The number of radionuclides with medical applications is extensive and increasing. In the areas of diagnosis and therapy, the most commonly used radiopharmaceuticals include chromium-51; cobalt-57, -58, and -60; gallium-67 and -68; technetium-99m; iodine-123, -125, and -131; selenium-75, xenon-127 and -133; and thallium-201. Biomedical researchers employ tritium, carbon-14, phosphorus-32, and sulfur-35 extensively. The radiopharmaceuticals used in medical applications may be obtained from radiopharmaceutical manufacturers or independent radiopharmacies, or they may be produced on site from radiopharmaceutical generators. Because of the relatively short half-lives of the radionuclides used in medicine, shipments from vendors are received frequently (weekly or daily), and storage times are minimal.

Radiopharmaceuticals purchased from vendors may be in the form of pre-packaged dose kits, radiopharmaceutical generators, or bulk supplies from which individual doses are extracted and prepared. Handling of prepackaged dose kits may involve no more than removing the material from the package and administering the radiopharmaceutical to the patient either orally or by intravenous injection.

Handling of materials obtained in the form of bulk stocks or radiopharmaceutical generators is more involved. In general, these materials are received and stored in a central area where individual doses are prepared. In the case of liquids, dose preparation involves extracting the required quantity from the stock solution by syringe or pipette and diluting the material in a suitable sterile medium. These operations are conducted in a fume hood, and the dose is administered to the patient either intravenously or orally.

Preparation of doses from radiopharmaceutical generators, of which molybdenum-99/technetium-99m generators are the most common, involves elution of the product from the generator and division of the elute into individual doses. The procedures for eluting a generator depend on whether it is a wet or dry column design. In a wet column generator, an evacuated extraction vial is attached to the end of the generator column with a sterile needle. Using the vacuum within the vial, the solvent is pulled from the generator reservoir through the column and into the vial. The procedure for a dry column generator is similar. However, since dry generators do not have a reservoir of solvent, solvent must be added to the column prior to elution. The charge vial is attached to one end of the generator, and then the evacuated extraction vial is attached to the other end. The solution is drawn through the generator column and collected in the elution vial. These elution procedures and dose divisions are

conducted in a fume hood, with the generator shielded to prevent external irradiation of the technicians.

Handling of radionuclides for biomedical research is more varied than that of radiopharmaceuticals used for patient administration. Depending on the specific radionuclides used and the goal of the experiment, the materials may simply be extracted from bulk stocks and administered, or the radionuclide may be subjected to additional chemical or physical processing.

2.2.1.2 Radiopharmaceutical Producers and Suppliers

Radiopharmaceutical manufacturers produce the radionuclide-labeled compounds, diagnostic kits, and radionuclide generators used in biomedical research and medical diagnosis and therapy. The radiopharmaceutical products may be shipped directly to medical users, or they may be shipped to independent radiopharmacies where individual doses are prepared from the bulk supplies or generators and distributed to medical users. Individual radiopharmaceutical manufacturers may specialize in only a few widely used radiopharmaceuticals or may produce many of the radionuclides used in biomedical research and patient diagnosis and therapy.

The radionuclides used in radiopharmaceuticals are produced either in nuclear reactors or in accelerators. Radiopharmaceutical manufacturers may operate their own production facilities or may purchase the bulk radionuclides from an outside vendor. In producing the bulk radionuclides, a suitable target is first prepared and then bombarded with neutrons or positive ions in the reactor core or accelerator. Once irradiation is complete, the target is removed from the production device, and the product is recovered and purified in a hot cell by appropriate chemical processing.

The production of the labeled compounds used in radiopharmaceuticals and biomedical research is essentially a wet chemistry process. Depending on the specific radiopharmaceutical, workers conduct these operations within laboratory fume hoods or gloveboxes. The final products are generally assembled and packaged in assembly line operations.

Radiopharmaceutical generators are designed and produced as closed aseptic systems using some type of chromatographic column. Typically, this chromatographic column consists of an inorganic ion exchange resin to which the generator (parent) radionuclide is bound. As the parent radionuclide decays, the decay product, which has different chemical/physical properties, is produced. The decay product is eluted from the column by the user at specified intervals. Generators are manufactured in a hot cell, where the parent radionuclide is packed in the column, and the column of the generator is surrounded by absorbent materials and shielding. The absorbent materials minimize the consequences of accidental breakage; the shielding reduces the radiation exposure of users. Once the generator is loaded, final assembly and packaging are carried out on an assembly line.

Independent radiopharmacies are a relatively recent phenomenon. Generally located in large cities, these facilities serve as distribution facilities. Radiopharmacies purchase bulk stocks and generators from radiopharmaceutical manufacturers and provide hospitals and clinics with individually prepared doses on an as-needed basis. The dose preparation procedures at these facilities do not differ from those at medical facilities that obtain their radiopharmaceuticals directly from the manufacturers.

2.2.2 Sealed Source Manufacturers

While facilities that use only sealed radiation sources are not covered by the NESHAP, the industrial facilities that produce sealed sources are subject to the standard. The facilities described in this section fall into two broad classes: those that manufacture encapsulated alpha, beta, or gamma-emitting radiation sources; and those that manufacture self-luminous devices.

2.2.2.1 Manufacturers of Sealed Radiation Sources

Sealed radiation sources are widely used in medical, industrial, and residential applications. Medical applications include gamma-emitting devices used in diagnostic and therapeutic procedures and sources used in patient implants. Industrial applications include nondestructive imaging and inspections, static eliminators, industrial gages, irradiation devices, and well-logging devices. The main radionuclides used in these devices iridium-142, krypton-85, americium-241, cesium-137, and cobalt-60. Smoke detectors, using alpha-emitting americium-241 sources, are the most widely used sealed sources in residential applications.

The manufacture of sealed sources is essentially a repackaging and redistribution process. Bulk radionuclides, in the form of pellets or foils, are received from a vendor in an approved shipping package. The shipping package is opened, and the required quantity of the radioactive material is removed and transferred to a container. The container is then sealed by welding or brazing. Most such devices are double encapsulated; i.e., an inner capsule contains the radioactive material and an outer container protects the inner container. Double encapsulation increases the assurance of safe handling. The outer container may also be brazed or welded, or simply screwed shut. All operations are performed in hot cells to protect the workers.

At some facilities, the bulk material purchased from the vendor is subjected to physical and/or chemical processing to alter the form of the material prior to encapsulation. For example, most cobalt-60 sources contain cobalt in the form of metal foils or microspheres. The cobalt is received from the vendor in the form of cobalt metal, and the material is processed by heating the metal to the melting point in a fluidizing furnace to form the desired microspheres. Similarly, manufacturers of smoke detectors generally obtain the bulk americium-241 in the form of oxide powder. This powder is compacted to form wafers, sintered in an induction furnace, ground to specifications, and hot-rolled with gold foil to produce the encapsulated material for incorporation into the device.

2.2.2.2 Manufacture of Self-Illuminating Devices

Self-illuminating devices include watches, compasses, signs, and aircraft instrumentation. Historically, radium-226 was used in radio-luminescent products. However, the well-documented hazards of working with radium and the advent of other materials with inherently superior characteristics have largely eliminated the use of radium. Today, tritium and, to a much lesser extent, krypton-85 and promethium-147 are used in the production of self-luminous devices.

Two general types of self-illuminating devices are made: those in which the radio-luminous material is incorporated into a paint which is used to coat the dial and/or instrument hands; and those in which a radioactive gas (tritium or krypton) is contained in a phosphor-coated glass ampule.

Manufacturers of self-illuminating devices obtain the bulk radio-nuclides in either gaseous or (rarely) liquid form from a vendor.

In the case of devices incorporating self-luminous paint, the manufacturing process involves the incorporation of the radionuclide in the paint and the application of the paint to the device. In the case of self-illuminating sources, the gaseous radionuclide (tritium or krypton-85) is transferred to the glass ampule and sealed. Both processes are carried out in areas with high ventilation rates or in fume hoods to protect the workers.

2.2.3 Test and Research Reactors

The NRC licenses approximately 70 academic, research, and industrial facilities to operate test and research reactors. Test and research reactors are used as teaching devices, to study reactor designs, to conduct research on the effects of radiation on materials, and to produce radioactive materials used by sealed source and radiopharmaceutical manufacturers.

The design of such reactors and their sizes vary widely. Approximately 15 research reactors are used primarily as teaching devices and have very low power outputs (less than 15 watts). The nuclear cores of these reactors have their uranium fuel dispersed and fixed in a plastic matrix. Given the design and use of these teaching reactors, airborne releases cannot occur during normal operations.

Research and test reactors used for experimental and production purposes include both light-water pool and heavy-water tank-type designs, ranging in power from 100 kilowatts to 10 megawatts. All of these facilities use highly enriched uranium fuel, either in metal or mixed carbide fuel elements.

In these reactors, experiments and/or production activities are conducted by remotely inserting the target containing the

material to be irradiated into the experimental ports or beam holes that penetrate the reactor core. The target material is subjected to the neutron flux of the reactor core for an appropriate period of time and then withdrawn via shielded transport devices (called "rabbit systems") to a hot cell. The irradiated material is examined or the product is recovered in the hot cell. Product recovery may be as simple as dissolving a soluble salt in water, or it may involve evaporation, precipitation, extraction, distillation, and/or ion exchange.

Potential airborne releases from such facilities include the fission products in the core of the reactor, activation products generated during the operation of the reactor, and releases from the disassembly and recovery of target materials in the hot cell. In general, the activation products, along with any gaseous fission products escaping the coolant, are released directly to the atmosphere from the facility exhaust. Materials that become airborne during processing in the hot cell will be vented through the hot cell's exhaust system. The effluent from the hot cell is generally filtered through high efficiency particulate air (HEPA) filters before release.

2.2.4 Non-Light-Water Reactor Fuel Fabricators

Only a few facilities produce the metal and mixed carbide fuel used in test and research reactors.

The non-oxide fuel fabrication process begins with highly enriched uranium metal. The uranium metal may be mixed with an alloying metal in an induction furnace. The fuel is then either rolled, punched, drilled, or crushed and compacted, and machined and shaped into the proper dimensions. Once the fuel is properly formed, it is enclosed in aluminum or stainless steel. The

enclosing process may involve injection casting, loading into a can or mold, or simply covering the fuel with side plates and rolling the metals together. Finished fuel elements are then inspected and cleaned prior to assembly into fuel bundles.

The production of mixed carbide fuel starts with highly enriched uranium dioxide-thorium dioxide powder ($\text{UO}_2\text{-ThO}_2$). This powder is mixed with graphite and heated to form uranium-thorium carbide kernels. These kernels are formed into microspheres by heating to a temperature in excess of the kernels' melting point. The microspheres are then coated with carbon and silicon layers in a fluidized bed furnace. Fuel rods are formed by injecting the coated kernels and a matrix material into a hot mold. The finished rods are then inserted into a graphite block to form the final fuel assembly.

2.2.5 Source Material Licensees

Two types of facilities are included in the category of "Source Material Licensees" which is subject to the NESHAP: those involved in the extraction of metals from uranium- and thorium-bearing ores, and those using depleted uranium metal or thorium in various products.

Approximately 10 facilities are engaged in the recovery of metals from source materials. In general, the products extracted from the uranium- and thorium-bearing ores are refractory metals, their oxides (columbium/niobium, zirconium, tantalum, and hafnium), or the rare earths (cerium, neodymium, dysprosium, etc.). These extraction operations involve processes typical of metal mining and beneficiation. Depending upon the specific facility and the products under recovery, the processing may involve wet chemical or solvent extraction, smelting, and high temperature sintering.

Facilities that manufacture products incorporating source materials include munitions producers using depleted uranium in armor-piercing projectiles, manufacturers that make lanterns and gas lights using thorium mantles, aerospace manufacturers using depleted uranium for stabilizers and ballast, and welding rod manufacturers that use thorium in the metallic form. Such manufacturers generally receive the material in the physical form in which it is used (e.g., depleted uranium in the form of metal billets). The processing is confined to such metallurgical operations as casting, forging, machining, and polishing.

2.2.6 Waste Receivers/Shippers and Disposal Facilities

The radioactive wastes generated by facilities that use radio-nuclides must be disposed of in an approved manner. In general, wastes with high specific activities (such as uranium-contaminated scrap at non-oxide fuel fabrication facilities) will be recycled and recovered. However, virtually every user of unsealed radioactive materials will generate solid, low-level radioactive wastes which require active disposal. Such wastes may be incinerated on site or packaged and shipped off site to a licensed low-level waste disposal facility.

Waste receivers and shippers (sometimes called "waste brokers") are primarily collection and shipping agents for facilities generating low-level wastes. Most such receiving/shipping facilities simply collect the wastes in shipping containers approved by the Department of Transportation from a number of waste generating facilities, monitor the packages for contamination, and hold the wastes at a warehouse until they arrange a shipment to a licensed disposal site. The licenses of most such receiving and shipping facilities do not allow the facility to repack or even open the waste packages. However, several such

facilities have been licensed to open, compact, and repackage waste materials before shipment.

Currently, there are three low-level radioactive waste disposal facilities which are accepting shipments for burial: the Barnwell facility in South Carolina, the Beatty facility in Nevada, and the Richland facility in Washington. Waste shipments are checked for damage and contamination upon receipt and then placed in excavated trenches. When a burial trench is filled with waste it is backfilled with soil.

2.3 URANIUM FUEL CYCLE FACILITIES

The uranium fuel cycle includes uranium mills, uranium hexafluoride conversion facilities, uranium enrichment facilities, light-water reactor fuel fabricators, light-water power reactors, and fuel reprocessing plants. With the exception of the uranium enrichment facilities that are owned by the Federal government and operated by contractors under the supervision of the Department of Energy (DOE), these facilities are licensed by the Nuclear Regulatory Commission (NRC) or the Agreement States.

2.3.1 Uranium Mills

Uranium mills extract uranium from ores which contain only 0.01 to 0.3 percent U_3O_8 . Uranium mills, typically located near uranium mines in the western United States, are usually in areas of low population density. The product of the mills is shipped to conversion plants, where it is converted to volatile uranium hexafluoride (UF_6) which is used as feed to uranium enrichment plants.

As of December 1988, of 27 uranium mills in the United States licensed by the NRC or agreement states, four were operating, eight were shut down, 14 were being decommissioned, and one had been built but never operated. The eight shut down mills could resume operations, but the 14 mills that are being decommissioned will never operate again.

The operating mills have a capacity of 9,600 tons of ore per day. The number of operating mills is down considerably from 1981, when 21 mills were processing approximately 50,000 tons of ore per day. This reduction reflects the decrease in the demand for yellowcake. The mined ore is stored on pads prior to processing. Crushing and grinding and a chemical leaching process separate the uranium from the ore. The uranium product is dried and packaged following recovery from the leach solution. The waste product (mill tailings) is piped as a slurry to a surface impoundment area (tailings pile).

Radioactive materials released to the air during these operations include natural uranium and thorium and their respective decay products (e.g., radium, lead, radon). These radionuclides, with the exception of radon, are released as particulates.

2.3.2 Uranium Conversion Facilities

The uranium conversion facility purifies and converts uranium oxide (U_3O_8 or yellowcake) to volatile uranium hexafluoride (UF_6), the chemical form in which uranium enters the enrichment plant.

There are currently two commercial uranium hexafluoride (UF_6) production facilities operating in the United States, the Allied Chemical Corporation facility at Metropolis, Illinois and the

Kerr-McGee Nuclear Corporation facility at Sequoyah, Oklahoma. The Allied Corporation facility, a dry-process plant in operation since 1968, has a capacity to produce about 12,600 mt of uranium per year in the form of UF_6 . The Kerr-McGee Nuclear Corporation facility is a wet-process plant in operation since 1970, with a capacity of about 9,100 mt per year (AEC74, Do88).

Two industrial processes are used for uranium hexafluoride production, the dry hydrofluor method and the wet solvent extraction method. Each method produces roughly equal quantities of uranium hexafluoride; however, the radioactive effluents from the two processes differ substantially. The hydrofluor method releases radioactivity primarily in the gaseous and solid states, while the solvent extraction method releases most of its radioactive wastes dissolved in liquid effluents.

2.3.2.1 Dry Hydrofluor Process

The hydrofluor process consists of reduction, hydrofluorination, and fluorination of the ore concentrates to produce crude uranium hexafluoride. Fractional distillation is then used to obtain purified UF_6 . Impurities are separated either as volatile compounds or as a relatively concentrated and insoluble solid waste that is dried and drummed for disposal.

2.3.2.2 Solvent Extraction Process

The solvent extraction process employs a wet chemical solvent extraction step at the start of the process to prepare high purity uranium for the subsequent reduction, hydrofluorination, and fluorination steps. The wet solvent extraction method separates impurities by extracting the uranium from the organic

solvent, leaving the impurities dissolved in an aqueous solution. The raffinate is impounded in ponds at the plant site.

2.3.3 Fuel Fabrication Facilities

Light water reactor (LWR) fuels are fabricated from uranium which has been enriched in U-235. At a gaseous diffusion plant natural uranium in the form of UF_6 is processed to increase the U-235 content from 0.7% up to 2% to 4% by weight. The enriched uranium hexafluoride product is shipped to LWR fuel fabrication plants where it is converted to solid uranium dioxide pellets and inserted into zirconium alloy (Zircaloy) tubes. The tubes are fabricated into fuel assemblies which are shipped to nuclear power plants. There are seven licensed uranium fuel fabrication facilities in the United States which fabricate commercial LWR fuel. Of the seven, only five had active operating licenses as of January 1, 1988. Of those five facilities, two use enriched uranium hexafluoride to produce completed fuel assemblies and two use uranium dioxide. The remaining facility converts UF_6 to UO_2 and recovers uranium from scrap materials generated in the various processes of the plant.

The processing technology used for uranium fuel fabrications consists of three basic operations: (1) chemical conversion of UF_6 to UO_2 ; (2) mechanical processing including pellet production and fuel-element fabrication; and (3) recovery of uranium from scrap and off-specification material. The most significant potential environmental impacts result from converting UF_6 to UO_2 and from the chemical operations involved in scrap recovery.

2.3.4 Nuclear Power Facilities

As of December 1986, there were 100 operable nuclear power reactors in the United States, with a total generating capacity of 85,177 MWe. With only one exception (a high temperature gas cooled reactor), all of these nuclear power reactors are either boiling water reactors (BWR) or pressurized water reactors (PWR). Pressurized water reactors comprise approximately two-thirds of the light-water generating capacity.

A light water-cooled nuclear power station generates electricity using the same basic principles as a conventional fossil-fueled (oil or coal) power station except that the source of heat used to produce steam is provided by nuclear fission instead of combustion.

In a boiling water reactor, the coolant boils as it passes through the reactor. The resulting steam is passed through a turbine and a condenser. The condensed steam is then pumped back into the reactor. The energy removed from the steam by the turbine is transformed into electricity by a generator.

The process is the same in a pressurized water reactor except that the reactor coolant water is pressurized to prevent boiling. Energy is transferred through a heat exchanger (steam generator) to a secondary system where the water does boil. Reactor coolant water is kept at high pressures by maintaining a closed system and electrically heating water in a tank called the pressurizer. After passage through the steam generator, the water is returned to the reactor. Secondary steam turns the turbine, is cooled in the condenser, and is pumped back into the steam generator.

During the fission process, radioactive fission products are produced and accumulate within the nuclear fuel. In addition,

neutrons produced during fission interact within the fuel and coolant to produce radioactive activation products. A reactor may experience periodic fuel failure or defects which result in the leakage of some of the fission and activation products out of the fuel and into the coolant. Accordingly, a typical light water reactor will experience build-up of radioactive fission and activation products within the coolant. For both PWRs and BWRs the radioactive contaminants which accumulate within the coolant are the source of radioactive emissions from the facility.

2.4 DEPARTMENT OF DEFENSE FACILITIES

The Department of Defense (DOD) operates a number of facilities that use unsealed sources of radioactive materials. In addition to three research and test reactors and numerous medical facilities, these include army bases that perform research and evaluation of munitions using depleted uranium and naval shipyards that service the Navy's nuclear-powered fleet.

The army bases that conduct research and development of munitions using depleted uranium metal are licensed by the NRC. Activities conducted at these facilities involve test firings and evaluations of various experimental and stockpile depleted uranium munitions such as armor piercing shells. At facilities performing research and development, activities can include the small-scale fabrication of depleted uranium projectiles. This fabrication can include forging, shaping, and grinding of depleted uranium metal.

Nine naval shipyards construct, refuel, maintain, and overhaul the submarines and ships of the Navy's nuclear-powered fleet: Mare Island Naval Shipyard in Vallejo, CA; General Dynamic's Electric Boat Division, Groton, CT; Pearl Harbor Naval Shipyard,

Pearl Harbor, HI; Portsmouth Naval Shipyard, Kittery, ME; Ingalls Shipbuilding Division, Pascagoula, MI; U.S. Naval Station and Naval Shipyard, Charleston, SC; Newport News Shipbuilding and Drydock Co., Newport News, VA; Norfolk Naval Shipyard, Portsmouth, VA; and Puget Sound Naval Shipyard, Bremerton, WA.

In addition to the normal shipyard functions of construction, maintenance and overhaul, these shipyards construct, test, refuel, and maintain the pressurized water reactors used to power the nuclear fleet. The primary source of radioactive emissions at naval shipyards is from the facilities that process and package radioactive wastes. These facilities handle solid low-level radioactive wastes such as contaminated rags, paper, filters, ion exchange resins, and scrap materials. Waste materials are sorted, surveyed, and packaged for shipment to disposal sites.

All effluent air systems at waste handling facilities are monitored during operation and equipped with HEPA filters. Environmental monitoring at these waste handling facilities indicates that the concentration of activity in the effluent air is actually lower than the background activity in the intake air (RI82).

REFERENCES

- AEC74 U.S. Atomic Energy Commission, Fuels and Materials Directorate of Licensing, Environmental Survey of the Uranium Fuel Cycle, April 1984.
- AM86 Amersham Corporation, "Products and Services for the Life Sciences," Arlington Heights, IL, 1986.
- BR84 Bremer, P.O., "Pharmaceutical Form--Packaging," Chapter 4 in Safety and Efficiency of Radiopharmaceuticals, Martinus Nijhoff Publishers, Boston, 1984.
- CE79 Cehn, J.I. et al., A Study of Airborne Radioactive Effluents From the Radiopharmaceutical Industry, Teknekron, Inc., McLean, VA, 1979.
- CO81 Cook, J.R., A Survey of Radioactive Effluent Releases From Byproduct Material Facilities, NUREG-0819, U.S. Nuclear Regulatory Commission, Washington, DC, 1981.
- Do88 Dolezal, W., personal communication with D. Goldin, SC&A, Inc. September 1988.
- FDA86 U.S. Food and Drug Administration, "Abbreviated Summary of Approved Radiopharmaceutical Drug Products," Washington, DC, 1986.
- GA86 Gallagher, B., Ph.D., New England Nuclear, personal communication, June 1986.
- ICN ICN Biomedicals, Inc., "ICN Radiochemicals," Irvine, CA, no date.

- KN84 Knapp, F.F., and Butler, T.A., editors, "Radionuclide Generators: New Systems for Nuclear Medicine Applications," ACS Symposium Series 241, American Chemical Society, Washington, DC, 1984.
- MA85 Mallinckrodt, Inc., "Product and Physical Data -- Radiopharmaceuticals," St. Louis, MO, 1985.
- ME86 Medi-Physics, "Technical Product Descriptions," Richmond, CA, 1986.
- ME78 Medical Economics Company, Litton Division, Physician's Desk Reference for Radiology and Nuclear Medicine 1978-9, Oradell, NJ, 1978.
- ME83 Merck Company, Merck Index, 1983.
- NEN New England Nuclear, "Radiopharmaceuticals and Nuclear Medicine Sources," North Billerica, MA, no date.
- NEN85 New England Nuclear, "Research Products 1985-6," North Billerica, MA, 1985.
- NEN83 New England Nuclear, "Sources and Accessories for Nuclear Medicine," North Billerica, MA, 1983.
- NRC78 U.S. Nuclear Regulatory Commission, Radioactivity in Consumer Products, NUREG/CP-0001, Washington, DC, 1978.
- RA83 Rayudu, G., editor, Radiotracers for Medical Applications: Volumes 1 and 2, CRC Press, Inc., Boca Raton, FL, 1983.

- RE86 Reba, Richard C., M.D., Director of George Washington University Nuclear Medicine Division, personal communication, June 1986.
- RI82 Rice, P.D., Sjoblom, G.L., Steele, J.M., and Harvey, B.F., Environmental Monitoring and Disposal of Radioactive Wastes from U. S. Naval Nuclear-Powered Ships and Their Support Facilities, Report NT-82-1, Naval Nuclear Propulsion Program, Department of the Navy, Washington, DC, 1982.
- SQ85 Squibb Diagnostics, "Technical Product Descriptions," New Brunswick, NJ, 1985.
- SU84 Sutter, S.L. et al., Emergency Preparedness Source Term Development for the Office of Nuclear Material Safety and Safeguards-Licensed Facilities, NUREG/CR-3796, prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Washington, DC, 1984.
- TU80 Tubis, M. and Wolf, W., Radiopharmacy, John Wiley and Sons, New York, 1980.
- US85 U.S. Pharmacopeia and the National Formulary, Washington, DC, 1985.

CHAPTER 3. DETERMINATION OF EMISSIONS

3.1 INTRODUCTION

This chapter provides background on the methods approved by the Agency for use by the owners or operators of NRC-licensed and non-DOE Federal facilities in determining the quantities of radioactive materials emitted to the atmosphere.

Assessment of the doses resulting from the release of radioactive materials into the air begins with a determination of the source term. Since dose is partly a function of the chemical and physical forms of the radionuclides in the effluent, a fully defined source term includes the emission rate and the physical characteristics of each chemical species in the effluent.

However, to demonstrate compliance with the standard, the source term need not be determined in such detail. In lieu of requiring the complex analytical procedures necessary to determine the exact composition of the source term, the approved procedures for determining compliance (see Chapter 4) incorporate dosimetrically conservative assumptions (i.e., assumptions that maximize the dose) about the physical characteristics and chemical forms of the radionuclides in the effluent. These conservative assumptions allow the source term to be defined simply by the emission rate for each radionuclide.

As discussed in Section 3.4, emissions may be calculated based on monitoring data obtained using EPA-approved sampling and analytical methods. However, since many of the facilities covered by the NESHAP do not have monitoring data on the quantities of radionuclides released to the atmosphere, the Agency has derived airborne emission factors and effluent control adjustment factors applicable to the facilities covered by the NESHAP. An airborne

emission factor is defined as the quantity of the material released to the air per unit time divided by the quantity of the material handled in an unsealed form over the same period of time. The owner or operator of a facility covered by the NESHAP may apply these EPA-approved emission factors and effluent control adjustment factors to the quantities of radioactive materials handled annually in unsealed form to determine the facility's emissions.

The remainder of this chapter is organized as follows: Section 3.2 describes the basis for the EPA-approved emission factors and presents these factors and their limitations; Section 3.3 presents the EPA-approved effluent control adjustment factors and the basis for the values assigned to each type of control; and Section 3.4 presents the basis for the EPA-approved sampling and analytical methods that may be used to calculate emissions.

3.2 EMISSION FACTORS

The fraction of material released to the atmosphere depends on the physical and chemical form of the radionuclide and the handling that the material undergoes. By analyzing the currently available emissions data for a range of radionuclides in various physical forms and undergoing differing processing/handling combinations, the Agency has derived generic "upper-limit" emission factors which, within the restriction given, can be used to estimate emissions conservatively. The approved emission factors apply to uncontrolled releases. Thus, when controls are used, credit can be taken for the control efficiency by applying the effluent control adjustment factors presented in Section 3.3.

Section 3.2.1 describes the approaches taken to obtain data and to derive the emission factors. Section 3.2.2 presents the approved emission factors and the limitations on their use.

3.2.1 Derivation of Emission Factors

The Agency conducted an extensive search for data suitable for deriving emission factors. Such data include reported release fractions, measured emissions, measured air concentrations, and measured intakes of radionuclides by workers. The approaches taken to obtain data are discussed in Section 3.2.1.1. The derivation of emission factors from each category of data is discussed in Section 3.2.1.2.

3.2.1.1 Procedures for Obtaining Data

Approaches used to obtain relevant data for defining airborne emission factors include:

- A keyword search of the following bibliographic data bases:
 - Pollution Abstracts (includes articles from the journal Health Physics);
 - International Pharmaceuticals (contains bibliographic references to articles on pharmaceutical chemistry and pharmacy practices);
 - The National Technical Information Service (NTIS) Data Base (contains references to government-sponsored reports);

- Health Planning and Administration (contains references on managerial practices in the medical field); and
- Environmental Bibliography (contains general environmental references).

The search of the primary bibliographic data bases used combinations of the keywords "airborne emissions," "radioactive waste," "radiopharmaceutical," "hospital," and "medical."

- The Public Document Room File Classification System of the NRC was searched using relevant keywords. This system is an automated mechanism for searching NRC-generated documents, meeting records, correspondence, etc.
- Representatives of a number of hospitals were contacted to collect information on the quantities of radionuclides emitted to the air. The hospitals that were contacted are: Washington University Medical Center; Georgetown University Hospital; George Washington University Hospital; Howard University Hospital; National Institutes of Health; Washington Hospital Center; Veterans Administration Hospital in Washington; Bethesda Naval Hospital; Children's Hospital; Holy Cross Hospital; Montgomery County Hospital; and Suburban Hospital.
- Representatives of the NRC's Material Licensing Branch were contacted in order to obtain relevant information from NRC personnel who license facilities subject to the NESHP. In addition, three

NRC inspectors (in NRC Regions I, II, and III) were contacted in an attempt to gain air emission data obtained during inspections.

- Representatives of the State Health Departments in New York, Pennsylvania, Virginia, and Maryland and the Health Department in the District of Columbia were contacted in order to collect air emission monitoring data and radionuclide release estimates for hospitals. In addition, a representative of the Conference of Radiation Control Program Directors was contacted in an attempt to collect relevant information. The membership of the Conference comprises all directors of radiation control programs in the 50 States, U.S. Territories, and some large municipal agencies.
- Members of the Council on Radiopharmaceuticals within the Society of Nuclear Medicine were contacted to obtain technical information concerning radiopharmaceuticals (e.g., physical and chemical forms and extent of use) and to identify possible sources of information on airborne releases from medical facilities.
- A printout of a data base maintained by the NRC's Office for Analysis and Evaluation of Operational Data was obtained and reviewed. This data base is a clearinghouse for written incident reports made to the Commission. In particular, reports from hospitals involving accidental releases of radioactive material from 1980 through 1985 were reviewed.

In total, over 70 useful books, and hundreds of reports, publications, and pieces of correspondence pertaining to the handling and airborne release of radionuclides were obtained through this search. After a preliminary screening of these materials, references that included emission factors, emissions data, air concentrations, or accidental intakes were reviewed in depth to determine if they included enough information to be useful in deriving emission factors.

3.2.1.2 Analysis of Data

Analysis of information relating to the derivation of emission factors varied depending on whether the data were reported as release fractions, airborne emissions, airborne concentrations, or worker intakes. The following subsections present the data obtained and describe the methods and assumptions used to derive emission factors from each of these types of data. Table 3-1, presented at the end of this section, summarizes the emission factors derived from the literature review. The entries in the table are keyed to the references discussed below.

Emission Factors Based on Reported Release Fractions

To be useful, information on the physical form of the material and the handling or processing associated with the reported release fraction is needed. Unfortunately, since the NRC's regulations are based on Maximum Permissible Concentrations, such direct measurements are rarely made. Only two references were obtained which provided release fractions based on measurements (CE79, EI83).

In A Study of Airborne Radioactive Effluents from the Radiopharmaceutical Industry (CE79), the airborne emissions from two unnamed hospitals were monitored: one for xenon-133 and the other for iodine-131.

The emissions of xenon-133 from a patient administration procedure was simulated by loading xenon in a spirometer bell jar. The spirometer was subsequently vented through a wall duct which connected to the main hospital exhaust system. A grab sample was taken from within the exhaust system and analyzed for its xenon content. No emission controls were present in the exhaust system between the point of release and the point where the sample was taken. The experiment was repeated four times. Based on the flow rate of the exhaust system, an emission factor of $4E-1$ was calculated.

At the second hospital, emissions of iodine-131, resulting from two separate patient applications of sodium iodide in a laboratory, were measured. The patient administrations took place over a 41-minute period. Samples were taken over a period of 81 minutes to span the administration period and a prudent additional amount of time. No effluent controls were present between the point of release and the point where the sample was taken. Based on the volume of air sampled, the emission factor for iodine-131 was $1E-5$.

"The Fraction of Material Released as Airborne Activity During Typical Radioiodinations" (EI83) describes the results of more than 150 release fractions calculated for iodine-125 used in research at the Washington University Medical Center. The calculated release fractions are based on the measured iodine trapped on activated charcoal samplers, the fumehood and air sampler flow rates, and the activity used in the experiments. The experiments involved variations of the chloramine-T, lactoperoxidase, and

iodine monochloride radioiodination techniques. They reflect activities in all of the facility's research laboratories. The mean emission factor is $8\text{E-}4$. The reference includes a statistical analysis of the data which indicates that the average emission factor from a series of 10 radioiodination procedures would not exceed $1\text{E-}3$ at the 99 percent confidence level.

Emission Factors Derived from Emissions Data

In order to derive emission factors from emissions data, it is necessary to know the quantity of material that is being processed, the form of the material and the processing or handling that it receives, and the types and efficiencies of any effluent controls used to reduce emissions.

The amount of emissions data available was limited because the NRC does not apply uniform licensing criteria concerning emissions monitoring and reporting to the facilities subject to the NESHAP. As a result, most of the facilities subject to the NESHAP are not required to perform effluent monitoring, and most of the facilities that are required to monitor emissions are not required to report these data. Even when emissions data are reported, the additional information needed to derive emission factors are infrequently provided.

The NRC conducted a voluntary survey of its licensees in 1980 to determine the quantities of radionuclides handled and releases of these materials to air and to water. The resulting data base, published as NUREG-0819 (CO81), includes responses from 50 percent of the NRC-licensed facilities handling unsealed radioactive materials.

Unfortunately, the survey did not request information on how emissions were determined, the physical forms of the radionuclides used and released, or the effluent controls in use. The lack of information on physical forms and effluent controls renders the data base unsuitable for developing the pre-effluent control emission factors of interest. Moreover, examination of the individual data points suggests that many of the reported airborne emissions are only rough estimates rather than measured values. The crudeness of many estimates is evidenced in cases where releases exceeding the total quantity of material handled are reported. Thus, the postcontrol emission factors that can be calculated from the data are highly suspect. Because of these inherent limitations, the data in NUREG-0819 were not used to determine emission factors.

Only two references were obtained that had sufficient information to derive emission factors (AL82, BR87). The first was the "Application for Renewal of Source Materials License: SUB-526, Docket 40-3392" for Allied Chemical UF₆ Conversion Plant (AL82). The Allied Chemical plant converts yellowcake to uranium hexafluoride. The handling and processing of materials in this facility is typical of facilities covered by the NESHAP. The process includes roasting and sizing the yellowcake feed received in 55-gallon drums from uranium mills, contacting the U₃O₈ feed with cracked ammonia to reduce it to UO₂, reacting the UO₂ with hydrogen fluoride to form UF₄, and reacting the UF₄ with fluorine to form uranium hexafluoride (UF₆). The UF₆ product is recovered in cold traps and distilled to yield the pure UF₆ product.

The application for renewal of the Allied Chemical Plant's source material license contains detailed airborne emissions data for the years 1979 through 1981. These emissions are given by individual release points. The application also contains detailed information on the effluent controls used on each release point

and the process served by each release point. This information and the knowledge that the plant was operating at capacity during this period made it possible to calculate emission factors for the various steps in the uranium conversion process.

Roasting and Sizing

The beneficiation of the crude yellowcake involves roasting and sizing the yellowcake to provide a uniform feed. The 3-year average emissions from this handling are 50 kg/yr uranium. Since 12,700 metric tons of U as UF_6 are processed annually, this is equivalent to a post-treatment emission factor of $4E-06$, assuming no loss of uranium in the process.

All of the release points for this process step are equipped with two baghouse filters in series. Assuming an operational efficiency of 99.5 percent for these filters (see Section 3.3), a pretreatment emission factor of $8E-04$ is derived.

Reduction

The uniform yellowcake feed is contacted with cracked ammonia to yield uranium oxide (UO_2). The 3-year average emissions from reducing the feed are 17.9 kg/yr U. Since no controls are used on the release points for the reducing step of the process, the emission factor calculated for this step is less than $2E-06$.

Hydrofluorination

In the hydrofluorination step the UO_2 is contacted with hydrogen fluoride to yield UF_4 , or greensalt. Only release points that handle the effluent from the hydrofluorination process with no effluent controls are considered. The 3-year average releases from the hydrofluorination blowers are 46.5 kg/yr U, equivalent to an emission factor of $4E-06$.

Fluorination

In the fluorination process, greencake feed (UF_4) is reacted with fluorine to produce the volatile uranium hexafluoride (UF_6) product. The 3-year average emissions from the fluorination process are 78 kg/yr U, equivalent to a post-treatment emission factor of $6\text{E-}06$. All release points from this processing step are treated to remove fluorine, hydrogen fluoride, and UF_6 . Treatment includes spray towers, packed towers, and metal filters. Assuming that these systems have an efficiency of 99.5 percent, a pre-treatment emission factor of $1\text{E-}03$ is calculated. Note that at this step, the feed has been converted to the hexafluoride form, a volatile solid.

In making the calculations, it was necessary to assume the quantity of the throughput and to assign efficiencies to the effluent controls. Since the plant was known to be operating at full capacity during the period, the annual throughput was assumed to be 12,700 metric tons of U as UF_6 . The assigned removal efficiency of 99.5 percent for the effluent controls is based on information on fabric filters and sintered metal filters presented in Chapter 2 of Control Technology for Radioactive Emissions to the Atmosphere at U.S. Department of Energy Facilities (MO84).

The second source of detailed emissions data was a personal communication from a large manufacturer of sealed sources (BR87). The manufacturer provided posteffluent control emissions data (Ci/yr) and throughputs for 1985, 1986, and the first 6 months of 1987 for a facility handling batches of cobalt-60 pellets containing several kilocuries of activity. The facility manufactures sources used in radiation therapy. The process involves transferring quantities of the cobalt pellets to the individual sources in a hot cell. The pellets have a high level of surface contamination, and dusting is evident during the transfer process.

Posteffluent control emission factors, calculated by dividing the measured releases by the quantity throughput, were calculated to be $1\text{E-}13$, $1\text{E-}11$, and $1\text{E-}8$ for each of the 3 years, respectively. Assuming an efficiency of 99.5 percent for the single-stage HEPA filter (see Section 3.3), pretreatment emission factors of $2\text{E-}11$, $2\text{E-}9$, and $2\text{E-}6$ are calculated.

While the increase in these emission factors over the period could not be definitively explained, it appears that the emissions data for the later years include a significant contribution from residual contamination in the hot cell.

Emission Factors Derived from Measured Concentration Data

In order to derive emission factors from measured concentration data, it is necessary to know the activity of the radionuclides present, the volumetric flow in the vent or area where the samples were taken, and the efficiency of any effluent controls between the point where the release occurs and the point where the samples are obtained. For noncontinuous releases (such as patient administrations), it is also necessary to know the duration of the process or procedure. The physical form of the material and the handling it undergoes must be known to characterize the circumstances associated with the release.

While concentration data are abundant in the literature, only five references provided enough of the additional data to derive emission factors (BNH80, BR78, EA80, LU80, WA87).

Air sample reports were prepared for three administrations of iodine-131 under normal conditions to patients at the Bethesda Naval Hospital (BNH80). The hospital's radiation safety staff

prepared the reports to help demonstrate compliance with the NRC's maximum permissible concentration (MPC) for iodine-131.

The iodine-131 was administered orally as sodium iodide in a sodium chloride solution. Two diagnostic doses and one therapeutic dose were given. Following each administration, the amounts of iodine-131 in samples of the room air were measured, and each value was divided by the volume of the air sample and corrected for the efficiency of the sampling filter to yield the air concentration.

To calculate the quantity of iodine-131 released to the air in each case, these reported air concentrations were multiplied by an air flow rate of $0.3 \text{ m}^3/\text{sec}$ and a time duration appropriate for such patient administrations. The quantity released was divided by the quantity administered to derive emission factors of $3\text{E}-7$, $2\text{E}-5$, and $5\text{E}-5$ for the three cases.

The air flow rate of $0.3 \text{ m}^3/\text{sec}$ is based on a review of flow rates in hoods venting directly to the atmosphere (NCRP89). The time duration for each case was assumed to be 10 minutes, a typical period when preparing and administering doses of iodine (BR78).

The second useful reference described an investigation of occupational exposure resulting from handling millicurie quantities of iodine-131 (BR78). These experiments measuring airborne radioactivity released from liquid and capsule forms of iodine-131 were conducted at the Monongahela Valley Hospital in North Charleroi, Pennsylvania.

A vial containing liquid (3-6 milliliters and 100-145 millicuries) iodine-131 was uncapped for 10 minutes to simulate typical

conditions for treatment preparation and administration. The experiment was conducted four times. The experiment was repeated six times with the vial left open for 1 hour to investigate the effect of time on the quantity of material released. In each experiment, the air concentrations of iodine-131 were measured by a sampler placed 4 feet from the vial. After correcting the measured activity for the efficiency of the sampler (78 percent), the air concentrations were calculated by dividing the measured activity by the total volume of air that passed through the air sampler.

Three similar experiments were conducted using iodine-131 in capsule form (the exact chemical form of the iodine was not reported in the study, but iodine-131 in capsules is typically in the form of sodium iodide). In each case, a vial containing 5 capsules of iodine-131 (approximately 20 millicuries each) was opened and exposed to the air for 1 hour. The resulting airborne concentrations were determined as described above.

The reported concentrations were multiplied by an assumed air flow rate of $0.3 \text{ m}^3/\text{sec}$ (based on NCRP89) and the length of time the vial was left open to determine the total quantities of iodine-131 released to the air. The quantity released was divided by the initial quantity present to derive the emission factors.

For the vials containing liquid and left open for 10 minutes, the iodine emission factors ranged from $3\text{E-}7$ to $9\text{E-}5$, with an average of $2\text{E-}5$. For the vials with liquid left open for 1 hour, the emission factors were roughly a factor of two higher, ranging from $6\text{E-}6$ to $2\text{E-}4$, with an average of $4\text{E-}5$. No increase in airborne activity was found in one of the experiments with the capsules. From the air concentrations for the other two experiments, the derived emission factor was $9\text{E-}7$.

The article "Monitoring of Airborne Contamination During the Handling of Technetium-99m and Radioiodine" describes testing at the Western Infirmary, Glasgow (EA80). The facility provides radiopharmaceutical and radiochemicals for hospitals and handles large activities of technetium-99m, iodine-125, and iodine-131. The assembly of technetium-99m radiopharmaceutical generators, technetium-99m generator elution, and the division of iodine-125 and iodine-131 stocks were monitored to determine airborne radioactivity.

Airborne radioactivity resulting from the daily generation of technetium-99m by Mo-99 generators in a fume cupboard was monitored inside the cupboard. The sampler was placed within 100 millimeters of the work site and operated throughout the elution procedure. Airborne releases from technetium-99m dose kit assembly (i.e., the incorporation of technetium-99m into a range of radiopharmaceuticals) were also monitored. Mean airborne concentrations, representing 10 measurements of the elution procedure and 20 measurements of the dose assembly procedure, and the duration of each procedure were reported.

Airborne releases from the division of stocks of iodine-125 solutions and the preparation of therapeutic doses from stocks of iodine-131 solutions were also monitored. While both are in the form of sodium iodide, the iodine-125 solution is acidic while the iodine-131 solution is formulated to be basic. Division of the iodine-125 stock involves opening a small screw-capped vial containing the solution of sodium iodide and transferring aliquots with a pipette to other vials. During the preparation of the iodine-131 doses, the sodium iodide solution is transferred from rubber-capped vials with a syringe to smaller rubber-capped vials. Mean air concentrations, representing 10 measurements of each of these activities, and the duration of each activity were reported.

In order to estimate the quantity of technetium-99m, iodine-125, and iodine-131 released to the air during each procedure, the mean concentrations were multiplied by an assumed air flow of 0.3 m³/sec (NCRP89) and the duration of the procedure. The quantities released were divided by the initial activities present to derive the following emission factors : 1E-5 for technetium-99m generator elution; 7E-6 for the Tc-99m dose assembly procedure; 2E-3 for the division of the iodine-125 stock; and 2E-5 for the preparation of iodine-131 doses.

The next reference describes a comparison study of radioiodine volatility from formulations of sodium iodide oral solutions used to treat thyroid cancer (LU80). This study was performed at the Tripler Army Medical Center in Honolulu, Hawaii. One of the formulations contained an oxidant (sodium bisulfite) intended to decrease the volatility of the iodine in solution.

Vials containing the iodide solution (80 to 176 millicuries) were uncapped and vented in a hood for 5 to 7 minutes where flowing air removed any airborne iodine to an exhaust vent on the roof. Air samples were collected from the rooftop vent using a constant-flow air sampler, and the resulting airborne concentrations of iodine-131 were reported.

From the airborne concentrations reported for the rooftop vent, an assumed air flow rate of 0.3 m³/sec (NCRP89), and the maximum duration of 7 minutes for venting the vials, the total quantity of iodine-131 released to the air was calculated for each of nine ventings. Dividing each of the quantities released by the initial quantity yielded the following emission factors. For the five ventings involving iodine-131 solution not containing sodium bisulfite, the emission factors ranged from 3E-4 to 1E-3 with a mean value of about 9E-4. The emission factors for the four

ventings involving the reformulated sodium iodide solution were about a factor of 70 lower, ranging from $2\text{E-}6$ to $2\text{E-}5$, with a mean of $1\text{E-}5$.

The final reference actually describes a study conducted to determine the feasibility of developing data-based guidelines for deciding which specific radiation protection measures should be used for operations involving unsealed radioactive materials (WA87). Although the study focuses on protection of workers, one of the purposes was to develop generic "dispersibility coefficients." The authors' definition of dispersibility coefficients is identical to the definition of emission factors used in this report. In the absence of site-specific data, the authors recommend dispersibility coefficients of 1 for gases, $1\text{E-}3$ for powders, $1\text{E-}4$ for water-based liquids, and $1\text{E-}6$ for solids subjected to sawing, grinding, sanding, or polishing.

The study also presents emission factors derived by the authors from data collected during site visits to nine facilities. Of these, two reflect pre-effluent control releases. For the forging and filing processes at a facility manufacturing nuclear fuel elements, an emission factor of $1\text{E-}7$ is reported. For the incinerator area, where uranium contaminated wastes are burned, the reported emission factor is $5\text{E-}7$.

Emission Factors Derived from Data on Worker Intakes

Measured accidental intakes of radionuclides by workers represent the last category of data used to derive emission factors. While accidental intakes can occur through skin contamination, ingestion, or inhalation, only intakes by inhalation were considered. When such data are reported in terms of the inhaled fraction of the activity handled (IFTAH), they can be used to derive the emission factors of interest.

In deriving the emission factors from IFTAH data, it is necessary to know the fraction of material released that is inhaled by the worker. Since this quantity is not directly measured, it had to be estimated theoretically. An upper limit to this fraction, 0.01 (derived in Appendix B), was used in all of the emission factor estimates based on measured intakes.

Because of the manner in which the fraction 0.01 was derived, only IFTAH data on accidental intakes were used to derive emission factors. Two references present IFTAH data for accidents (DO66, FR68).

Donth and Maushart present information involving accidental intakes from 14 incidents during a 4-year period at the Karlsruhe Nuclear Research Center in Karlsruhe, Germany (DO66). Franke et al., via a literature survey, collected data on 20 accidents from which IFTAH data were derived (FR68). These 20 accidents include the 9 incidents reported by Donth and Maushart, where enough information was available to determine IFTAH.

The 20 accidents reported by Franke et al. were reviewed to determine the activity that led to the accidental intake and to ensure that no process containment or effluent controls were present which would have reduced the potential exposure of the worker. Fifteen of the reported accidents, ranging from a broken container to explosions in gloveboxes, were deemed suitable for deriving emission factors.

Six emission factors for radionuclides in liquid form were derived from the IFTAH data: three for liquids at ambient temperatures and three for liquids at elevated temperatures. For liquids at ambient temperatures, the emission factors range from $2\text{E-}6$ to $2\text{E-}5$. The emission factors for liquids at elevated temperatures range from $3\text{E-}4$ to $1\text{E-}3$. Seven emission factors were

derived for radionuclides in powder forms. For powders at ambient temperatures, the six derived emission factors range from $1\text{E-}6$ to $7\text{E-}5$. The emission factor for the one accident involving a powder at elevated temperatures is $1\text{E-}3$. Finally, two emission factors were derived for solids: $3\text{E-}6$ for a solid at ambient temperatures, and $2\text{E-}4$ for a solid which was being brazed at 900°C .

Table 3-1. Summary of Reported and Derived Emission Factors

Radio-nuclide	Description of Process or Activity	Emission Factor	Reference
MATERIALS IN GASEOUS FORM			
U-nat	Production and recovery of UF ₆	1E-3	AL82
Xe-133	4 simulated patient dose administrations	4E-1	CE79
MATERIALS IN LIQUID FORM AT AMBIENT TEMPERATURES			
I-131	Patient dose administration	1E-5	CE79
I-131	Patient dose administration	3E-7	BNH80
I-131	Patient dose administration	2E-5	BNH80
I-131	Patient dose administration	5E-5	BNH80
I-131	4 simulated patient dose administrations with vials left open 10 minutes	2E-5	BR78
I-131	6 experiments with vials left open for 1 hour	4E-5	BR78
I-131	10 dose preparations using a syringe to transfer solution from a rubber-capped vial to smaller rubber-capped vials (solution formulated to reduce volatility)	7E-6	EA80
I-131	5 simulated dose preparations using solutions formulated to reduce volatility	1E-5	LU80
I-131	4 simulated dose preparations using solutions not buffered to reduce volatility	9E-4	LU80
I-125	150 measurements in clinical research laboratories during radioiodination procedures	8E-4	EI83
I-125	10 dose preparations pipetting solution from an open vial to smaller open vials (solution not formulated to reduce volatility)	2E-3	EA80
MFP	Accidental rupture of container in a laboratory	5E-5	FR68
Sr-90	Accidental spilling in a laboratory	2E-6	FR68
Tc-99m	10 elutions from Mo-99/Tc-99m generators	1E-5	EA80
Tc-99m	20 preparations of Tc-99m Dose Kits	7E-6	EA80
Zr/Nb-95	Accidental spilling in a laboratory	4E-5	FR68

Table 3-1. Summary of Reported and Derived Emission Factors (continued)

Radio-nuclide	Description of Process or Activity	Emission Factor	Reference
MATERIALS IN LIQUID FORM AT ELEVATED TEMPERATURES			
MFP	Accidental splashing during distillation in a laboratory	7E-4	FR68
MFP	Accidental splashing during distillation in a laboratory	1E-3	FR68
Ru-106	Evaporation of sample in a laboratory	3E-4	FR68
MATERIALS IN POWDER FORM AT AMBIENT TEMPERATURES			
Cs-137	Accidental rupture of container in a laboratory	2E-5	FR68
Pa-233	Accidental rupture of container in a laboratory	2E-5	FR68
Pu-238	Glovebox explosion	1E-5	FR68
Pu-239	Glovebox explosion	1E-6	FR68
Sr-90	Accidental rupture of window in a laboratory	4E-5	FR68
Sr-90	Broken container in storeroom	7E-5	FR68
U-nat	Conversion of UO ₂ to UF ₄	4E-6	AL82
MATERIALS IN POWDER FORM AT ELEVATED TEMPERATURES			
Eu-152	Accidental spill while heating in a hood	1E-3	FR68
U-nat	Roasting and sizing of yellowcake	8E-4	AL82
MATERIALS IN SOLID FORM AT AMBIENT TEMPERATURES			
Co-60	Manufacture of sealed sources in a hot cell	2E-9	BR87
Pu-239	Explosion of container during cryogenic operation in laboratory	3E-6	FR68
U-nat	Forging and filing of uranium billets	1E-7	WA87
MATERIALS IN SOLID FORM AT ELEVATED TEMPERATURES			
U-nat	Incineration of uranium-contaminated scrap	5E-7	WA87
Cs-137	Brazing of materials at 900° C in a laboratory	2E-4	FR68

3.2.2 Approved Emission Factors

The approved emission factors presented in Table 3-2 were determined from the information given in Section 3.2.1. They were developed separately for gases, liquids and powders, and solids and capsules.

These emission factors may not be used for estimating emissions from uranium mill tailings piles. Instead, use the methods given in NRC87.

3.2.2.1 Gases

The radionuclides used in gaseous form at the facilities subject to the NESHAP include argon, carbon-14, krypton, tritium, and xenon. Additional radionuclides such as sulfur, bromine, and uranium may also exist in gaseous forms. Krypton and xenon are used primarily at medical facilities, while carbon-14 and tritium are widely used in research facilities. Gaseous tritium is also used to manufacture self-illuminating signs and instruments.

The approved emission factor for all radionuclides in gaseous form is 1.0. It is recognized that this emission factor will overestimate the actual emissions in many cases. Clearly, an industrial facility using gaseous tritium to produce self-illuminating signs or a radiopharmaceutical manufacturer producing xenon for patient administration would not have any product to sell if all of the gaseous tritium or xenon were released to the air. While only two emission factors were derived in Section 3.2.1 for gases, the emission factor of $4E-1$, measured during a simulated patient administration of xenon, indicates that the fractional release can indeed approach unity. Thus, based on the intrinsic properties of gases, the manner in which they are used

Table 3-2. EPA-Approved Emission Factors

Physical Form-Processing	Emission Factor
Gases*-All processes	1.0
Liquids or Powders-Processes involving temperatures in excess of 100° C or intentional dispersion	1.0
Any radionuclide, regardless of physical form that is intentionally released into the environment	1.0
Liquids or Powders-Any process not involving either temperatures in excess of 100° C or intentional dispersion	1E-3
Solids-Processes involving temperatures in excess of 100° C	1.0
Solids**-Any process not involving temperatures in excess of 100° C	1E-6
Mo-99 in Mo-99/Tc-99m generators	1E-6

* Any material with a boiling point of 100° C or less shall be considered a gas for the purpose of determining the appropriate emission factor.

**Capsules containing radionuclides in liquid or powder form can be considered to be solids.

in medical and research applications, and the limited amount of data available, an emission factor of 1.0 for gases has been selected.

3.2.2.2 Liquids and Powders

Almost all of the radionuclides used by the facilities subject to the NESHAP can be obtained in the form of liquids or powders. Radionuclides used for medical purposes are almost always in the liquid form for ease of administration. Similarly, radionuclides used in research are most frequently in the form of liquids or powders to facilitate handling. Facilities that manufacture radiopharmaceuticals handle materials in liquid form, as do the facilities that produce self-illuminating watches and other products using tritiated paints. Producers of smoke detectors and other products using radioactive materials in a solid or encapsulated form may receive the material in powdered form and convert it to a solid.

The emission factors derived in Section 3.2.1 for liquids range from $3\text{E-}7$ to $2\text{E-}3$; those derived for powders range from $1\text{E-}6$ to $1\text{E-}3$. These emission factors represent more than 250 measured data points. They reflect such diverse activities as radiopharmaceutical dose preparation, patient dose preparation and administration, beneficiation of powdered ore concentrates, chemical processing and conversion, routine research activities, and laboratory accidents. Because of the similarity in the magnitudes of the emission factors derived for liquids and powders, a single emission factor of $1\text{E-}3$ is approved for both forms. This value is roughly equivalent to the maximum emission factor derived for either liquids or powders.

The background information on the emission factors given in Table 3-1 was reviewed to determine the limits of their applicability. Based upon this review, it was determined that some activities to which these radionuclides are subjected could result in emission factors greater than $1\text{E-}3$. The following paragraphs explain the basis of this conclusion and present restrictions on the application of the approved emission factor for liquids and powders.

The dispersibility of a material to the air is a complex phenomenon, depending on a number of physical and chemical parameters associated with the material and its processing. Two significant parameters associated with high releases are the volatility of the material and the temperature of the process. Although none of the derived emission factors for liquids and powders exceed the magnitude of the approved value, several of the values in Table 3-1 are roughly comparable. In particular, several of the derived emission factors for liquid solutions containing iodine in various chemical forms and some for materials at elevated temperatures are close to or roughly equivalent to $1\text{E-}3$.

It is well known that iodine in the elemental form is volatile. Although solutions that contain iodine used in research and medicine are generally buffered to reduce the formation of elemental iodine, some of the iodine may be converted to elemental iodine and released to the air. The extent to which this has occurred in the processes reviewed to derive the emission factors given in Table 3-1 is not completely known. However, the activities reviewed include a variety of medical and research uses of radioiodines, and the data reflect fractional releases from both buffered and nonbuffered iodine solutions. Thus, the approved emission factor of $1\text{E-}3$ is believed to be adequate for iodine solutions not subject to elevated temperatures. Lacking data for other volatile radionuclide solutions, the EPA has determined

that any material with a boiling point of 100° C or less shall be considered to be a gas (i.e., emission factor = 1) for the purpose of determining emissions.

The data for radionuclides processed at elevated temperatures are also insufficient to demonstrate the adequacy of the 1E-3 emission factor for all activities involving elevated temperatures. For this reason, the EPA is assigning an emission factor of 1 to all radionuclides that are heated to temperatures of 100° C or higher. Although this may be conservative, the Agency wants to be certain that there is almost no probability that the approved emission factors will be exceeded in practice.

Similarly, the data on which the approved 1E-3 emission factor is based do not cover activities in which radioactive material is deliberately dispersed in the environment. In the absence of information relating to such activities, the Agency is assigning an emission factor of 1 to any activity involving intentional dispersion of radionuclides in the environment.

There is also a special case in which an emission factor smaller than 1E-3 is approved for liquids and powders. This is for the use of molybdenum-99 generators to produce technetium-99m. As discussed in Chapter 2, these generators are widely used for medical applications. The NRC's regulations (10 CFR Section 35.14(b)(4)(iii)) stipulate that the amount of molybdenum-99 present in the technetium-99m elute that is administered to a patient cannot exceed 0.001 millicurie/millicurie technetium-99m. Since the elution of technetium-99m from the generator is essentially a closed system, the only molybdenum-99 which is available for release to the air is that which is contained in the elute. Since the physical decay characteristics of molybdenum-99 are such that only 87 percent of the molybdenum-99 in the generator can be eluted as technetium-99m, the fraction of molybdenum-99

contained in the eluted solution cannot exceed $9\text{E-}4$. Applying the approved emission factor for liquids to this maximum fraction in the elute, the result is an airborne emission factor for molybdenum-99 of $9\text{E-}7$. In practice, this value will be much lower, since most generators are designed to meet the more stringent purity requirements of the U.S. Pharmacopoeia, i.e., 0.15 microcurie of molybdenum-99 per millicurie technetium-99m. Given these considerations, the Agency will allow the emission factor for solids ($1\text{E-}6$) to be applied to molybdenum-99 in molybdenum-99/technetium-99m generators.

3.2.2.3 Solids

Relatively few of the facilities covered by the NESHAP use radioactive materials in solid or capsule forms. The most prevalent use of solid materials is in the manufacture of sealed radiation sources. A few radiopharmaceuticals, notably iodine-131, are available as capsules. If a liquid or powder is contained in a sealed capsule and is not exposed to a temperature exceeding 100°C , it may be considered to be a solid.

The approved emission factor for solids and capsules is $1\text{E-}6$. This value is consistent with the derived emission factors. It is valid for all radionuclide/handling processing combinations except those involving volatile solids or solids subjected to elevated temperatures. For solids with a boiling point of 100°C or less, the material shall be considered a gas, and an emission factor of 1 shall be used. For solid materials subjected to high-temperature processing, that is heated to 100°C or higher, an emission factor of 1 shall be used.

3.3 ADJUSTMENT FACTORS FOR EFFLUENT AIR CONTROL DEVICES

3.3.1 Use of Effluent Air Control Devices

The use of effluent controls can significantly reduce the quantities of radionuclides released to the environment. For example, based on the emission factor presented in Table 3-2 for liquids, air emissions for a hospital using 100 millicuries of iodine-131 in solution would be 100 microcuries in the absence of effluent controls. If, however, the material were vented through an exhaust system equipped with charcoal filters, much of the iodine would be removed before its release to the atmosphere, and the air releases would be less than 100 microcuries. Similarly, the releases from a facility handling 1 millicurie of americium oxide powder would be estimated to be 1 microcurie in the absence of effluent controls. However, if the material were handled within a glovebox equipped with a HEPA filter, the actual quantity of material released to the environment would be a small fraction of the material reaching the filter. Therefore, in estimating the airborne emissions, adjustment factors based on the types of emission controls used can be applied to the uncontrolled emission factors presented in Table 3-2.

This section describes the emission controls used at facilities covered by the NESHAP and characterizes their effectiveness. Approved adjustment factors and the rationale for their selection are provided for each type of control. These approved adjustment factors, which are summarized in Table 3-3, can be applied to the approved uncontrolled emission factors given in Table 3-2 to derive the emission rates needed to determine compliance.

Two classes of controls are used to control airborne releases at the facilities covered by the NESHAP. The first class comprises air flow or ventilation systems. The objective of these systems

Table 3-3. Approved Adjustment Factors for Effluent Controls

Control	Types of Airborne Radionuclides Controlled	Approved Adjustment Factor	Comments and Conditions
HEPA Filters	All particulate forms (e.g., technetium)	0.01/stage	Not applicable for gaseous radionuclides; periodic testing would be prudent to ensure high removal efficiency
Fabric Filters	Particulates	0.1	Monitoring would be prudent to guard against tears in filter
Sintered Metal Filters	Particulates	1	Insufficient data to make recommendation
Activated Carbon Filters	Iodine Gas	0.1	Efficiency is time dependent -- monitoring is necessary to ensure effectiveness
Douglas Bags:			
Held one week or longer for decay	Xenon	0.5/wk	Based on xenon half-life of 5.3 days;
Released within one week	Xenon	1	Provides no reduction to public exposures
Xenon Traps	Xenon	0.1	Efficiency is time dependent -- monitoring is necessary to ensure effectiveness

Table 3-3. Approved Adjustment Factors for Effluent Controls (continued)

Control	Types of Airborne Radionuclides Controlled	Approved Adjustment Factor	Comments and Conditions
Venturi Scrubbers	Particulates	0.1	Although venturis may remove gases, variability in gaseous removal efficiency dictates adjustment factor for particulates only
	Gases	1	
Packed-Bed Scrubbers	Soluble Gases	0.1	Not applicable for particulates or insoluble gases
Electrostatic Precipitators	Particulates	0.05	Not applicable for gaseous radionuclides

is to redirect the radionuclides away from the workers and to the outside atmosphere where dilution occurs. These controls, primarily unfiltered fume hoods and vent stacks, do not reduce the quantity of radionuclides released to the environment. Use of these types of controls does not affect a facility's emissions, and no adjustment factor is approved. The second class of controls physically trap and hold radionuclides, thus preventing their escape to the outside environment. Controls that fall into this class include filters, bags, and traps. Use of these types of controls reduce a facility's airborne releases.

3.3.1.1 HEPA Filters

HEPA filters contain continuous sheets of fiberglass paper pleated back and forth over corrugated separators housed in either a wooden or steel casing. Individual HEPA filters are designed for air flows of approximately 1,000 cfm but are frequently arranged in filter banks to handle larger air flows. HEPA filters are ideal for capturing particulate contaminants in effluent streams with moderate flow rates and dust loadings. They provide no protection against radionuclides emitted in gaseous forms (e.g., krypton and xenon).

The integrity and efficiency of HEPA filters can deteriorate if they are exposed to excessive moisture, heat, corrosive fumes, or vibration. These conditions might be expected in some industrial facilities, where harsh chemical processing operations take place and a wide variety of contaminants is emitted.

Although HEPA filters are rated at a minimum efficiency of 99.97 percent for 0.3 micrometer particulates, empirical data from the Savannah River Laboratory indicate that the in-service efficiency for a single-stage HEPA filter is approximately 99.5 percent

(SRP79). Moreover, to achieve this efficiency, a HEPA filter must be installed properly, maintained frequently, and replaced periodically. For this reason, a conservative adjustment factor of 0.01 is approved for single-stage HEPA filters. For each additional stage of HEPA filters mounted in series, an additional factor of 100 can be allowed. Thus, adjustment factors of 0.0001 and 0.000001 are approved for two-stage and three-stage HEPA systems, respectively.

3.3.1.2 Baghouse Filters

A baghouse filter (sometimes referred to as a fabric filter) is another high-efficiency particulate control system. The baghouse filter system consists of filter bags of felt (or a similar material) arranged in tubular fashion in an enclosed housing. The effluent stream is blown through the filter bags, which trap the particulates primarily on the collected material which builds up on the bags. As the buildup of material on the bags increases, resistance to flow increases. Thus, baghouse filters must be equipped with a shaking, vibrating, or reverse-flow device to remove a portion of the collected dust and recondition the bags.

Like HEPA filters, baghouses may be adversely affected by moisture, high temperatures, or corrosive materials in the effluent. Unlike HEPA filters, they are designed for high air flows and heavy dust loadings. The rated efficiency of a baghouse filter can be as high as 99.9+ percent. According to the data presented in MO84, efficiencies of 99.5 percent are typical for particulates in the respirable range of 10 microns or less. However, during actual operations, the efficiency will be lower due to ruptures in the bags and releases during bag reconditioning. Therefore, an adjustment factor of 0.1 is approved for baghouse filters.

3.3.1.3 Sintered Metal Filters

Sintered metal filters are designed for high-efficiency particulate removal when the temperature of the effluent air or corrosive contaminants in the effluent make HEPA and baghouse filters unsuitable. A sintered metal filter is constructed by fusing a mass of metal particles together under heat and pressure. Rated efficiencies are equivalent to HEPA and baghouse filters; i.e., greater than 99 percent. However, sintered metal filters are not commonly used at the facilities covered by the NESHAP. Moreover, since data on their operational efficiencies could not be obtained, an approved adjustment factor is not provided for these devices.

3.3.1.4 Activated Carbon Filters

Adsorption on activated carbon is used primarily to control gaseous emissions of radioactive iodine. Carbon filters are usually housed in fume hoods. The carbon is usually treated with triethylene diamine (TEDA) to guard against the retention of noble gases and to enhance the retention of organic radioiodine. Carbon filters are rarely used alone; they are usually preceded by particulate filters to remove materials that could load and clog the carbon.

The efficiency of a freshly mounted carbon filter frequently exceeds 99 percent (MO84). According to Cehn et al. (CE79), however, the removal efficiency may drop to as low as 90 percent prior to cartridge replacement. Factors that affect the efficiency include the type of charcoal used, the age of the filter, the iodine concentration in the discharge air, the air flow rate, the relative humidity, the temperature, the type of impregnants used, and the extent of iodine loading on the filter. In general,

replacement of spent carbon filters is required about once per year, although this frequency may vary depending on the chemical purity of the process air and other factors mentioned above.

Based on a conservative estimate of 90 percent for the removal efficiency, the approved adjustment factor for activated carbon filters is 0.10.

3.3.1.5 Douglas Bags

The Douglas bag is an inexpensive control device used to limit xenon releases during patient administrations. The Douglas bag is a single-use container designed to capture xenon exhaled from the patient. Following patient administration, xenon within a Douglas bag is either bled immediately (or within a short time) into a fume hood or held in a lead tank for radioactive decay before release.

The efficiency of a Douglas bag depends on the length of retention of the xenon before it is released to the atmosphere. The half-life of xenon-133, the principal isotope of xenon used in medical facilities, is 5.3 days. Therefore, if the xenon is held for about 5 days prior to release, almost half of the xenon decays.

Because retention times vary between hospitals and even between separate departments within the same hospital, it is not possible to approve a single value for the adjustment factor. If xenon is released from a Douglas bag in less than 1 week after the time of administration, no adjustment factor may be applied to the xenon emission factor. For each week of retention, the xenon emission factor may be adjusted by a factor of 0.5. For example, if xenon were retained for a period of 3 weeks, the adjustment factor would be $0.5 \times 0.5 \times 0.5 = 0.125$.

3.3.1.6 The Xenon Trap

The xenon trap is a relatively new but increasingly popular method of controlling xenon releases from hospitals. The system consists of a pipe filled with an activated carbon cartridge and an air pump which draws the contaminated air through the carbon filter. With xenon traps, the air may either be exhausted to the outside or recycled back to the clinical area where the administration takes place.

Cehn et al. (CE79) state that a xenon trap removes 98 percent of the xenon in an emission stream; Early et al. (EA85) cite a removal efficiency of 95 percent. Problems do occur, however, due to saturation of the carbon. Because it is quite difficult to determine saturation, frequent monitoring is necessary to ensure adequate performance. In addition, the effectiveness of the carbon cartridge may be reduced due to water-vapor retention. For protection against water vapor in a patient's exhalation, a silicagel cartridge is frequently placed in front of the carbon filter. The silica-gel cartridge requires weekly checks to ensure the effectiveness of the system.

Therefore, a conservative estimate of 90 percent is assigned to the efficiency of xenon traps to account for common problems in their performance. Accordingly, the approved adjustment factor for a xenon trap is 0.10.

3.3.1.7 Venturi Scrubbers

Venturi scrubbers are used primarily to control particulate material, although they may be moderately effective in controlling gases that are soluble or reactive with the scrubber solution (usually water). In a venturi scrubber, effluent air is

forced at high speed through a pipe restriction in which water is radially introduced. The drop in pressure of the effluent air caused by the restriction atomizes the water, and the water droplets effectively intercept particulates (CA84).

The particle removal efficiency of venturi scrubbers is reported to be approximately 95 percent. Their gas removal efficiency is quite variable (depending on the type of reaction and the time of contact between the effluent air and the scrubber solution).

Therefore, the approved adjustment factor is conservatively given as 0.1 for particulates, and no adjustment factor is provided for gases.

3.3.1.8 Packed-Bed Scrubbers

A packed-bed scrubber is a vessel filled with randomly oriented packing material such as ceramic rings, spirals, or saddles. A scrubbing liquid is fed to the top of this packed bed while the effluent air flows through the bed either concurrently, counter-currently, or crosscurrently with the scrubber solution. As the liquid flows through the bed, it wets the packing material, which traps gaseous emissions.

Packed-bed scrubbers can be highly efficient in removing gaseous material (sometimes more than 99 percent efficient) but are generally not effective in removing particulates (BO80). The factors affecting the efficiency of packed bed scrubbers include the depth of the packing material, the contact time of the effluent with the material, the solubility of the gaseous material, the drop in pressure of the effluent air as it passes through the packing material, the effluent air flow rate, and the type of scrubbing liquid used.

A conservative estimate of the efficiency of a packed bed scrubber in removing soluble gases is 90 percent. Therefore, the approved adjustment factor for soluble gases is 0.1. Since these systems are not designed remove small particulates, an adjustment factor for particulates is not given.

3.3.1.9 Electrostatic Precipitators

Electrostatic precipitators (ESPs) are costly but very effective devices for particulate removal. ESPs employ high voltages to impart a negative electrical charge to particles in the effluent. The particles are attracted to and collected on positively charged plates. ESPs usually provide higher removal efficiencies for particles smaller than 1 micron in diameter than do venturi scrubbers. Their particle removal efficiencies may exceed 97 percent (SK); however, the ESPs' performance is sensitive to the electrical resistivity and stickiness of the particles as well as to the particle size (BO80).

Considering these factors, the approved particulate adjustment factor for electrostatic precipitators is 0.05.

3.4 APPROVED SAMPLING AND ANALYTICAL METHODS

Facilities that wish to determine their emissions empirically may do so. However, the methods used to obtain and analyze samples must conform to the following requirements:

1. Effluent flow rate measurements (velocity and volumetric flow) from point sources shall be made

using: Reference Method 2 of Appendix A to 40 CFR Part 60 for stacks or large vents; or Reference Method 2A of Appendix A to 40 CFR Part 60 for pipes and small vents.

2. Effluent sampling points shall be selected using the criteria of Reference Method 1 of Appendix A to 40 CFR Part 60.
3. Representative samples of the effluent stream shall be withdrawn in accordance with the guidance presented in ANSI-N13.1, "Guide to Sampling Airborne Materials at Nuclear Facilities."
4. Radionuclides shall be collected and measured using procedures based on a principle of measurement equivalent to those described in Method 114 of Appendix B to 40 CFR Part 61.

The methods and principles of measurement given in Appendixes A and D of 40 CFR Part 60 are standard sampling and radioanalytical techniques that have been reviewed for suitability and are in common use.

If an owner or operator of an existing facility determines that it is impractical to sample an effluent stream in accordance with requirements 2 and 3, alternative procedures may be used provided: the reason the required methods are impractical is fully documented; the alternative procedures are fully documented; the alternative procedures will not cause emissions to be significantly understated; and the emissions do not cause doses in excess of 10 percent of the standard. The owner or operator of a facility which employs analytical methods not described in Appendix D must submit the methods to the Agency for review and approval prior to using them to determine emissions.

REFERENCES

- AL82 Allied Chemical UF₆ Conversion Plant, "Application for Renewal of Source Materials License: SUB-526, Docket 40-3392," Metropolis, IL, July 1982.
- BA83 Barrish, E.L., et al., "Radioactive Waste Management at Biomedical and Academic Institutions," Proceedings of an International Conference on Radioactive Waste Management Held by the IAEA In Seattle, WA, May 12-20, 1982, 1983.
- BNH80 Bethesda Naval Hospital, "Three Air Sample Reports Following Normal Iodinations at Bethesda Naval Hospital in September and October 1980," unpublished, provided by Capt. Charles Galley.
- BO80 Bonner, T., et al., Engineering Handbook for Hazardous Waste Incineration, Monsanto Research Corporation, prepared for the U.S. Environmental Protection Agency, Washington, DC, 1980.
- BR87 Brodsky, A., personal communication with a large manufacturer of sealed sources, 1987.
- BR78 Browning, E.J., Banerjee, K., and Reisinger, W.E., "Airborne Concentrations of I-131 in a Nuclear Medicine Laboratory," Journal of Nuclear Medicine, Vol. 19, pp. 1078-1081, 1978.
- BU76 Burchsted, C.A., et al., Nuclear Air Cleaning Handbook: Design, Construction, and Testing of High-Efficiency Air Cleaning Systems for Nuclear Application, ERDA 76-21, Distribution Categories UC-11, 70, Oak Ridge National Laboratory, Oak Ridge, TN, 1976.

- CA84 California Air Resources Board and the South Coast Air Quality Management District, Southern California Hazardous Waste Incineration: A Feasibility Study, Prepared for the Southern California Hazardous Waste Management Project, 1984.
- CE79 Cehn, J.I., A Study of Airborne Radioactive Effluents From the Pharmaceutical Industry, Final Report, Prepared by Teknekron, Inc., for the U.S. EPA Eastern Environmental Research Facility, Montgomery, AL, 1979.
- CO81 Cook, J., A Survey of Radioactive Effluent Releases From Byproduct Material Facilities, NUREG-0819, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC, 1981.
- DO66 Donth, H.H., and Maushart, R., "Empirical Radiotoxicity Hazards and General Licensing Exemptions," Health Physics, Vol. 12, No. 1, pp. 106-108, 1966.
- EA80 Eadie, A.S., Horton, P.W., and Hilditch, T.E., "Monitoring of Airborne Contamination During the Handling of Technetium-99m and Radioiodine," Phys. Med. Biol., Vol. 25, No. 6, pp. 1079-1087, 1980.
- EA85 Early, P.J., Principles and Practice of Nuclear Medicine, The C.B. Mosby Company, St. Louis, MO, 1985.
- EI83 Eichling, J., "The Fraction of Material Released as Airborne Activity During Typical Radioiodinations," Proceedings of the 9th Biennial Conference of Campus Radiation Safety Officers, University of Missouri at Columbia, June 6-8, 1983.

- EG83a EG&G Idaho, Inc., Improved Low-Level Radioactive Waste Management Practices for Hospitals and Research Institutions, Report No. DE83-015465, 1983.
- EG83b EG&G Idaho, Inc., Massachusetts Low-Level Radioactive Waste Management Survey, Report No. DOE/LLW-19T, 1983.
- EPA84 U.S. Environmental Protection Agency, Background Information Document (Integrated Risk Assessment), Final Rules for Radionuclides, Volume II, Report No. EPA 520/1-84-022-2, Washington, DC, 1984.
- FR68 Franke, T., Hermann, G., and Hunzinger, W., "A Quantitative Estimation of the Hazards Involved in Work with Radionuclides," in Radiation Protection - Proceedings of the First International Congress of Radiation Protection, Rome, 1966, Vol. 2., edited by W.S. Snyder, et al., Pergamon Press, London, pp. 1401-1406, 1968.
- FR75 Frost, D. and Jammet, H., Manual on Radiation Protection in Hospitals and General Practice, Volume 2, Unsealed Sources, World Health Organization, 1975.
- GR75 Gregory, W.D. and Maille, H.D., "Incineration of Animal Radioactive Waste--A Comparative Cost Analysis," Health Physics, 29 , pp. 389-392, September 1975.
- ICF86 ICF, "Hazardous Waste Incineration Feasibility Study for Vandenberg Air Force Base," prepared for Headquarters Space Division, Department of the Air Force, 1986.
- IL86 Illinois Department of Nuclear Safety, Annual Report on the Survey of Low-Level Radioactive Waste Generators in Illinois for 1984, 1986.

- LU80 Luckett, L.W., and Stotler, R.E., "Radioiodine Volatilization from Reformulated Sodium Iodide I-131 Solution," Journal of Nuclear Medicine, 21:477-479, 1980.
- MO84 Moore, E.B., Control Technology for Radioactive Emissions to the Atmosphere at U.S. Department of Energy Facilities, Report No. PNL-4621, Prepared for U.S. EPA Eastern Environmental Radiation Facility by Pacific Northwest Laboratory, Richland, WA, 1984.
- NCRP89 The National Council on Radiation Protection and Measurements, Screening Techniques for Determining Compliance with Environmental Standards: Releases of Radionuclides to the Atmosphere, NCRP Commentary No. 3, Revision of January 1989, Bethesda, MD, January 1989.
- NRC87 "Methods for Estimating Radioactive and Toxic Airborne Source Terms for Uranium Milling Operations," Regulatory Guide 3.59, U.S. Nuclear Regulatory Commission, March 1987.
- NY84 New York State, Low-Level Radioactive Waste Management Study, Volume Two, Main Report, 1984.
- OZ84 Oztunali, O. and Roles, G., De Minimus Waste Impacts Analysis Methodology, NUREG/CR-3585, U.S. Nuclear Regulatory Commission, Washington, DC, 1984.
- SK Skoski, L., et al., Airborne Radioactive Emission Control Technology, Volume III. Prepared by Dames and Moore for the Office of Radiation Programs, U.S. Environmental Protection Agency, Washington, DC, no date.

- SRP79 Lee, M. W., and Stoddard, D.H., Memorandum to J. T. Buckner, "Statistical Analysis of HEPA Filtration Systems," DPST-79-359, Savannah River Laboratory, Technical Division, duPont Company, Savannah, GA, May 1, 1979.
- HHS83 U.S. Department of Health and Human Services, "Workshop Manual for Radionuclide Handling and Radiopharmaceutical Quality Assurance," Food and Drug Administration, Public Health Service, Rockville, MD, 1983.
- WA87 Watson, C.E., and Fisher, D.R., Feasibility Study of a Data-Based System of Decisions Regarding Occupational Radiation Protection Measures, NUREG/CR-4856, prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Washington, DC, February 1987.

CHAPTER 4. COMPLIANCE PROCEDURES AND EXEMPTION CRITERIA

4.1 INTRODUCTION

The NESHAP limits the quantities of radioactive materials that can be released to the air annually to such quantities that will not result in any member of the public receiving a dose greater than the standard. Since the doses resulting from exposure to radioactive materials cannot be measured directly, they must be calculated based on the specific radionuclides involved, the magnitude of the exposure for each exposure pathway, and pathway-specific dose factors derived from dosimetric and metabolic models.

The significant exposure pathways for radionuclides released to the air are: air immersion, ground-surface contamination, inhalation, and ingestion. While the magnitude of the exposure from each of these pathways can, theoretically, be determined by environmental measurements, such measurements are difficult and costly. Moreover, as noted in Chapter 1, at the concentrations consistent with the limits of the standard, it may not be possible to distinguish the fraction that is due to the emission from the part due to background radioactivity.

The difficulties and costs associated with environmental measurements led to the development of mathematical models, such as the computer code AIRDOS-EPA, for estimating exposures. The actual exposure caused by the release of radioactive materials into the air is an extremely complex function. It depends on the kind and quantity of radionuclides released; the physical configuration of the facility releasing the materials; the dispersion, transport and build-up of the radionuclides in the soil and foodstuffs; and the proximity to the facility of individuals and farms producing foodstuffs.

To account for these factors, the more sophisticated models require extensive data inputs. For example, the AIRDOS-EPA code provides for the user to supply: radionuclide-specific release rates by lung clearance class; radionuclide-specific scavenging coefficients and deposition velocities; release point-specific data concerning release height, volumetric flow rate, and heat content of the effluent air; site-specific meteorological data including lid height, joint frequency distributions of wind by stability class, and annual precipitation rates; and site-specific demographic data including population distribution and the fraction of food produced within the assessment area.

In general, the more specific the input data, the more precise will be the estimate of exposure and dose. However, it is not necessary to determine precisely the doses that result from the emissions of a given facility to determine whether that facility is in compliance with the limit established by the NESHAP. Compliance will be unambiguously demonstrated when estimated upper-bound doses are below the limit of the standard. Furthermore, valid upper-bound dose estimates may be calculated using dosimetrically conservative assumptions in lieu of site-specific data for any of the parameters affecting dose.

The validity of upper-bound estimates has been widely recognized. In its 1984 report on radiological assessment, the National Council on Radiation Protection and Measurements (NCRP) recommended that "...the 'best' model for a given [dose] assessment will be the model which is easiest to use and which produces results within an acceptable degree of accuracy" (NCRP84). In its Commentary No. 3, "Screening Techniques for Determining Compliance with Environmental Standards: Releases of Radionuclides to the Atmosphere," the NCRP used this approach to develop a series of three "Screening Levels" for estimating upperbound doses (NCRP89). At each successive level, the dose estimate becomes

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more precise as additional user-supplied site-specific data and additional computations replace dosimetrically conservative parametric values that were assumed at the preceding level.

The EPA has approved the NCRP's Commentary No. 3 as a means of determining compliance with the NESHAP and has extended the approach to develop three additional approved compliance procedures. The use of conservative assumptions in the derivation of the compliance procedures ensures that the level of protection afforded the public by the limit of the standard is fully achieved. By providing these compliance procedures, the Agency will minimize the burden on licensees of determining compliance with the standard. In addition, the Agency has adopted exemption criteria to reduce the burden of the reporting and construction approval requirements of the NESHAP.

Section 4.2 discusses the basis for the models used in developing the approved procedures for estimating the dose from a given release. In Section 4.3, each of the procedures is briefly described in terms of the user inputs required and the dispersion models and other assumptions used in the assessment. Section 4.4 presents the Agency's criteria for determining exemption from the reporting requirements of the NESHAP.

4.2 MODELS USED IN THE COMPLIANCE PROCEDURES

4.2.1 Atmospheric Dispersion Models

The dose received from a given release of radioactive materials into the air depends on the exposure received through each of the significant environmental pathways. Calculating the magnitudes of these exposures first entails determining the concentration of the radionuclides in the air at the point where the most

exposed individual is located, and, if different, the concentration at the location where foodstuffs consumed by that individual are raised.

The air concentration at any point in the environment is a complex function of the quantity of the radioactive material released, the configuration of the facility from which the material is released, the distance from the point of the release to the locations of interest, the prevailing meteorological conditions, and various depletion processes which remove the radioactive material from the effluent plume as it moves from the point of release to the location of the receptor.

The simplest approximation of the air concentration for each of the exposure pathways is to assume the concentration is the same as the concentration at the point of release. Since it cannot be greater than the concentration at the point of release, the maximum concentration in the environment can be calculated by the following equation:

$$C = Q/V \quad (1)$$

where,

C = concentration (Ci/m^3),

Q = release rate of the radionuclide (Ci/s), and

V = the volumetric flow in the stack or vent from which the material is released (m^3/s).

The release rate (Q) is determined using the approved procedures presented in Chapter 3. Where the volumetric flow (V) in the

stack or vent is not known, the value of $0.3 \text{ m}^3/\text{s}$ may be used as a default. The NCRP recommends this default value, based on a conservative estimate of typical flows in fume hoods (NCRP89).

Equation 1 does not account for any dispersion of the material in the atmosphere. This assumption is unrealistic even when the receptor is close to the point of the release, as the wind does not always blow in the same direction. To account for this fact, equation 1 is modified to:

$$C = fQ/V \quad (2)$$

where,

f = the frequency of wind toward the direction of the receptor (dimensionless).

The frequency of wind is computed for each of 16 compass points. For all of the compliance procedures except one, the value of f is assumed to be 0.25, the value recommended by the NCRP. This value is conservative, as actual meteorological data sets rarely show a value greater than 15 percent ($f = 0.15$) in the predominate wind direction.

While equations 1 and 2 are appropriate for use in "upper-bound" dose estimates, they do not account for dilution of the concentration in the effluent plume as it travels from the point of release to the location of the receptor. Diffusion and atmospheric turbulence are the primary processes acting to reduce the concentrations in the plume. Secondary removal processes include gravitational fallout and wet and dry deposition.

The degree of dilution resulting from atmospheric turbulence and diffusion depends upon the stability of the atmosphere, the joint

frequency distribution of wind speed and direction, and the distance from the point of release to the location of the receptors. Additional factors that influence dilution include the height at which the release occurs, the rise of the effluent plume due to the momentum and/or thermal buoyancy of the gases in the effluent, and the relationship between the height of the release and the heights of the building from which the release occurs and surrounding structures.

The importance of each of these factors varies depending upon the physical configuration of the facility from which the release occurs and the locations of the receptors. Thus, several dispersion equations (given in NCRP86) are used in the compliance procedures to calculate air concentrations. For situations where the release point is isolated from the perturbed air flow caused by buildings, the dispersion is calculated based on Gifford's formulation of the ground-level centerline Gaussian plume equation (GI68):

$$C = \frac{fQ}{u \sigma_y \sigma_z} \exp \left[-1/2 \left(\frac{H}{\sigma_z} \right)^2 \right] \quad (3)$$

where,

u = mean wind speed (m/sec),

H = height of effluent release (m), and

y and z = the horizontal and vertical diffusion parameters (m).

The extent of horizontal and vertical diffusion is a function of the atmospheric stability and the distance (x) from the point of release to the point of the receptor. For the annual average air

concentrations of interest, neutral atmospheric stability is appropriate (GI75). Briggs (BR74) gives the horizontal and vertical diffusion parameters as:

$$\sigma_y = (0.08x) \left(\frac{1}{\sqrt{(1 + 0.0001x)}} \right) \quad (4)$$

$$\sigma_z = (0.06x) \left(\frac{1}{\sqrt{(1 + 0.0015x)}} \right) \quad (5)$$

In developing the compliance procedures, the sector-average formulation of the Gaussian model given in equation 6 is used for those situations where the point of release (H) is greater than 2.5 times the height of the building (H_p) from which the release occurs:

$$C = \frac{fQP}{u} \quad (6)$$

where,

P is the Gaussian diffusion factor given by Fields and Miller (FI80) as:

$$P = \frac{2.032}{x \sigma_z} \exp \left[-1/2 \left(\frac{H}{\sigma_z} \right)^2 \right] \quad (7)$$

Where H is greater than zero, the concentration is not simply an inverse function of the distance from the point of release to the location of the receptors; over some distance (dependent upon the height of release), the concentrations will be lower than the maximum ground-level concentration at the point where the plume

touches down. In developing the compliance procedures, the value of P at the point of maximum ground-level concentration is used over all such distances to ensure that the maximum exposure is evaluated.

In cases where buildings perturb the air flow, the Gaussian model may not apply. Three situations are considered in the development of the compliance procedures: the receptor is located in the same building from which the release occurs, the receptor is located within the near wake region (wake recirculation zone), and the receptor is located beyond the near wake region.

When the receptor is located in the same building from which the release occurs, the concentration at the point of the receptor is calculated according to the recommendations of Wilson and Britter (WI82) as follows:

$$C = B_0 \frac{Q}{u_h x^2} \quad (8)$$

where,

u_h = the mean wind speed (m/sec) in undisturbed air at the level of the building's roof, and

B_0 = an empirically derived constant that accounts for potential zones of stagnation along vertical surfaces due to building wakes.

In the procedures, B_0 is set equal to 30, which is the largest value given by Wilson and Britter. Such a value will occur when both the point of the release and the receptor are located on the lower third of the building.

For the situations where building wakes are a factor, but the receptor is not living in the building from which the release occurs, the appropriate dispersion model depends on whether the receptor is located in the wake recirculation zone (near wake). The extent of the wake recirculation zone is a function of the cross-sectional area of the building from which the release occurs. It is defined as including all distances x , where x is less than or equal to the square root of the projected building area. For this situation, the dispersion equation of Miller and Yildran is used (MI84):

$$C = \frac{fQ}{uh_bK} \quad (9)$$

where,

h_b = the smaller of the building height or width, and

K = a constant with the value 1 m.

Where the receptor is located outside the wake recirculation zone, the Gaussian model (equation 6) is modified by replacing the diffusion parameter P with a diffusion parameter B , which accounts for the reduced concentrations due to trapping and recirculation of the effluent within the near wake zone. Values of B for various building sizes and distances are presented in Figure 4 of NCRP's Commentary No. 3 (NCRP89).

Where plume rise due to momentum or buoyancy are considered they are calculated according to the equations provided by Briggs, and are defined in the User's Guide for COMPLY (EPA89b).

4.2.2 Models Used to Estimate Exposures

Once the ground-level air concentrations are calculated at the locations of interest, it is necessary to estimate the magnitude of the exposure received via each of the significant pathways. For each of the procedures except the "Compliance Model" of the COMPLY code, the doses resulting from exposures by each of these pathways are calculated in the manner described in NCRP's Commentary No. 3 (NCRP89).

For the "Compliance Model" (Level 4 of the COMPLY code) the methods are generally the same, but there are some differences. The contribution of daughter radionuclides to the dose from external exposure is handled internally by the computer program, rather than being built into the dose factors. Build-up of the daughters in the food chain is also calculated by the program instead of being compensated for in the dose factors. The differential equations in the ensuing discussion are solved using the methods given by Skrabale (Sk74).

The air concentration for estimating the exposure from immersion and inhalation is handled as follows. The concentration of the parent, N_1 , is calculated from:

$$dN_{A1}/dt = -\lambda_1 N_{A1};$$

where N_{A1} is the concentration of parent atoms, ($= C_{air}/\lambda_1$), and C_{air} is the concentration of the parent calculated from $C_{air} = \bar{Q}(C/Q)$.

The daughter concentration is,

$$dN_{Ai}/dt = N_{Ai-1} - \lambda_i N_{Ai}.$$

The initial condition is $N_i = 0$ at $t = 0$. The equations are solved for $t = T$, where T is the transport time from the source to the receptor ($= x/u$), where x is the distance and u is the average wind speed.

The concentration of decay products contributing to the dose from the exposure to contaminated ground is estimated as follows:

$$dN_{G1}/dt = vN_{A1} - (\lambda_1 + \lambda_{h1})N_{G1}$$

$$dN_{Gi}/dt = vN_{Ai} + \lambda_{i-1}N_{Gi-1} - (\lambda_i + \lambda_{h1})N_{Gi};$$

where v is the deposition velocity and λ_{h1} is the environmental removal constant. All the concentrations are initially set to zero, and the equations are solved for $t = 100$ years.

Grow-in of decay products in the food chain is accounted for as follows. The concentration of the leaves of plants is given by,

$$M dN_{L1}/dt = vA f_r N_{A1} - M(\lambda_1 + \lambda_w)N_{L1}.$$

Since $Y = m/A$, this becomes,

$$dN_{L1}/dt = (v f_r / Y) N_{A1} - (\lambda_1 + \lambda_w) N_{L1}.$$

The equations for the daughters are,

$$dN_{Li}/dt = (v f_r / Y) N_{Ai} + \lambda_{i-1} N_{Li-1} - (\lambda_i + \lambda_w) N_{Li},$$

where M is the plant mass growing on area A , f_r is the retention factor, λ_w the weathering constant, and Y the biomass per unit area at harvest. The initial concentrations are all zero. These differential equations are solved for the N 's at $t = t_e$, the length of the growing season. The N 's are converted to activity by the equation $C_i = \lambda_i N_i$.

The concentration in the soil available for uptake by the roots of the plants is given by,

$$M dN_{R1}/dt = AvN_{A1} - M(\lambda_1 + \lambda_{h1})N_{R1}$$

$$M dN_{Ri}/dt = AvN_{Ai} + \lambda_{i-1}N_{Ri-1} - (\lambda_i + \lambda_{h1})N_{Ri},$$

where M is the mass of soil in the root zone, A the surface area of soil on which the radioactivity deposits, v the deposition velocity, and λ_{h1} the removal constant for harvesting and leaching. Because P (the areal density of the root zone) is equal to M/A, the equations become,

$$dN_{R1}/dt = (v/P)N_{A1} - (\lambda_1 + \lambda_{h1})N_{R1}, \text{ and}$$

$$dN_{Ri}/dt = (v/P)N_{Ai} + \lambda_{i-1}N_{Ri-1} - (\lambda_i + \lambda_{h1})N_{Ri}.$$

The concentration in the plant is simply

$$C_i = B_{vi}(\lambda_i N_{Ri})$$

and the total concentration is the sum of the leaf and soil uptake concentrations. The equations are solved for $t = 100$ years for build-up, and the N's are converted to activity by

$$C_i = \lambda_i N_i.$$

After harvest, slaughter, or milking the parent and daughter nuclides behave according to

$$dN_1/dt = -\lambda_1 N_1$$

$$dN_i/dt = \lambda_{i-1}N_{i-1} - \lambda_i N_i$$

The initial values are the values at the end of the growing season. The times are the delay times for the vegetables, meat, and milk. Note that the concentrations are assumed to be constant from the time the animal eats the forage until it is milked or slaughtered.

These concentrations are used with the pathway equations in the NCRP's Commentary No. 3 (NCRP89) to calculate intake and dose.

Tritium and carbon-14 at Level 4 are treated slightly differently than in the NCRP's Commentary No. 3. The NCRP's approach assumes that the specific activity of carbon-14 and tritium are the same in the food as in the atmosphere. Level 4 uses a similar approach, described in Baker (Ba76). Instead of assuming that the specific activity of tritium is the same in the food product as in the atmosphere, the Baker method accounts for some dilution by non-tritiated water. In addition, Level 4 uses the equations given by Baker directly, rather than using transfer factors.

Wet deposition is treated as follows. According to the International Atomic Energy Agency (IA82), the washout factor, $W(1/m^2)$, is given by

$$W = Nc\rho_i/2\pi x_i u_i,$$

where N is the number of sectors, c is a factor equal to 1.8×10^{-5} yr/mm-sec for particulates and 1.2×10^{-5} yr/mm-sec for iodines, r_i is the rainfall rate in sector i , m/yr, x_i is the distance from the source to the receptor, and u_i is the annual average wind speed.

The "wet deposition velocity", V_w is defined as

$$V_w = A_w/C,$$

where A_w is the flux of the material deposited on the ground ($\text{Ci}/\text{m}^2\text{-sec}$) and C is the air concentration (Ci/m^3). The flux is given by

$$A_w = QW,$$

where Q is the annual average release rate.

Combining these equations,

$$V_w = A_w/C = (Q/C)Ncp_i/(2\pi x_i u_i).$$

The total deposition is then the sum of the wet and dry deposition velocities. The precipitation rate is taken to be 1 meter per year.

The only other difference between Level 4 and the other compliance procedures is the value of the parameters. The values used in Level 4 are those listed in Table 7-2 in Chapter 7 of Volume 1 of the Environmental Impact Statement prepared in support of the NESHAP (EPA89).

4.2.3 Derivation of Dose Conversion Factors

After calculating the magnitude of the exposure for each environmental pathway, the resulting doses to the organs of the body are calculated by applying appropriate radionuclide-specific dose factors for each pathway. For the compliance procedures, these dose factors were derived according to the dosimetric and metabolic models recommended by the International Council of Radiation Protection (ICRP) in Publication 30. The weighting factors recommended by the ICRP in Publication 26 are applied to the organ dose factors to derive the effective whole-body dose equiv-

alent. These dose factors are tabulated for each pathway in either Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion (EPA88) or External Dose Rate Conversion Factors for Calculation of Dose to the Public (DOE88).

4.3 APPROVED COMPLIANCE PROCEDURES

The compliance procedures developed for use by facilities covered by the NESHAP are described fully in EPA89a, and EPA89b. These procedures are based on the air dispersion, terrestrial transport, and dosimetric and metabolic models described in Section 4.2, and, in the case of the first two procedures, the emission factors developed by the Agency to estimate maximum radionuclide emissions. The procedures minimize both the amount of site-specific data that the user must provide and the computations that the user must make to determine compliance. In fact, all but one of the procedures can be performed using only paper and pencil and a hand calculator. The one procedure, the "Compliance Model" of the computer code COMPLY, that does require a personal computer to perform, is extremely easy to run. The other procedures can also be performed using the COMPLY computer code.

Each of the compliance procedures is described briefly in the following paragraphs.

4.3.1 Procedure 1: Quantity of Material Handled

The first compliance procedure is based on the quantities of radionuclides handled at the facility. The Agency has derived a

Table of Annual Possession Quantities for Environmental Compliance (see Table 4-1) which gives the quantity of each radionuclide that can be handled annually without causing any member of the public to receive a dose in excess of the limits of the standard.

In deriving the quantities in the table, the Agency assumed the most exposed individual resides 10 meters from the point of release and obtains food grown at a location 100 meters from the point of release. The most conservative model for treating the dispersion of radionuclides in the air was used to estimate concentrations at 10 and 100 meters from the release point, which was assumed to be at ground level. The quantity of each radionuclide that could be released without exceeding the limit of the standard was then calculated using the pathway-specific effective whole-body equivalent dose factors for each radionuclide. The annual possession quantity for each physical form was then calculated by applying the appropriate emission factors. Because of the assumptions made regarding the dispersion of the effluent, the procedure may be used only by facilities where no one resides within 10 meters of the release point and no food production occurs within 100 meters.

To apply this procedure, users at facilities meeting the restriction need only to:

- a. determine the quantity of each physical form of each radionuclide handled at the facility;
- b. compute for each physical form of each radionuclide the ratio of the quantity handled to the annual possession quantity given in Table 1 of Section 61.106; and
- c. sum the computed ratios.

Table 4-1 Annual Possession Quantities for Environmental Compliance

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
Ac-225	9.6E-05	9.6E-02	9.6E+01
Ac-227	1.6E-07	1.6E-04	1.6E-01
Ac-228	3.4E-03	3.4E+00	3.4E+03
Ag-106	1.6E+00	1.6E+03	1.6E+06
Ag-106m	2.6E-03	2.6E+00	2.6E+03
Ag-108m	6.5E-06	6.5E-03	6.5E+00
Ag-110m	2.4E-05	9.4E-02	9.4E+01
Ag-111	6.7E-02	6.7E+01	6.7E+04
Al-26	4.0E-06	4.0E-03	4.0E+00
Am-241	2.3E-06	2.3E-03	2.3E+00
Am-242	1.8E-02	1.8E+01	1.8E+04
Am-242m	2.5E-06	2.5E-03	2.5E+00
Am-243	2.3E-06	2.3E-03	2.3E+00
Am-244	4.6E-02	4.6E+01	4.6E+04
Am-245	7.0E+00	7.0E+03	7.0E+06
Am-246	9.8E-01	9.8E+02	9.8E+05
Ar-37	1.4E+06	-	-
Ar-41	1.4E+00	-	-
As-72	2.9E-02	2.9E+01	2.9E+04
As-73	6.0E-02	6.0E+01	6.0E+04
As-74	4.3E-03	4.3E+00	4.3E+03
As-76	8.8E-02	8.8E+01	8.8E+04
As-77	7.9E-01	7.9E+02	7.9E+05
At-211	1.0E-02	1.0E+01	1.0E+04
Au-193	4.2E-01	4.2E+02	4.2E+05
Au-194	3.5E-02	3.5E+01	3.5E+04
Au-195	3.3E-03	3.3E+00	3.3E+03
Au-198	4.6E-02	4.6E+01	4.6E+04
Au-199	1.5E-01	1.5E+02	1.5E+05
Ba-131	1.0E-02	1.0E+01	1.0E+04
Ba-133	4.9E-05	4.9E-02	4.9E+01
Ba-133m	9.3E-02	9.3E+01	9.3E+04
Ba-135m	5.8E-01	5.8E+02	5.8E+05
Ba-139	4.7E+00	4.7E+03	4.7E+06
Ba-140	2.1E-03	2.1E+00	2.1E+03
Ba-141	1.3E+00	1.3E+03	1.3E+06
Ba-142	1.1E+00	1.1E+03	1.1E+06
Be-7	2.3E-02	2.3E+01	2.3E+04
Be-10	3.0E-03	3.0E+00	3.0E+03
Bi-206	3.1E-03	3.1E+00	3.1E+03

See footnotes at the end of the table.

Table 4-1 Annual Possession Quantities for Environmental Compliance (continued)

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
Bi-207	8.4E-06	8.4E-03	8.4E+00
Bi-210	4.2E-03	4.2E+00	4.2E+03
Bi-212	4.7E-02	4.7E+01	4.7E+04
Bi-213	6.0E-02	6.0E+01	6.0E+04
Bi-214	1.4E-01	1.4E+02	1.4E+05
Bk-249	7.0E-04	7.0E-01	7.0E+02
Bk-250	1.0E-01	1.0E+02	1.0E+05
Br-77	7.5E-02	7.5E+01	7.5E+04
Br-80	1.2E+01	1.2E+04	1.2E+07
Br-80m	1.5E+00	1.5E+03	1.5E+06
Br-82	1.6E-02	1.6E+01	1.6E+04
Br-83	9.9E+00	9.9E+03	9.9E+06
Br-84	5.6E-01	5.6E+02	5.6E+05
C-11	1.3E+00	1.3E+03	1.3E+06
C-14	2.9E-01	2.9E+02	2.9E+05
Ca-41	2.7E-02	2.7E+01	2.7E+04
Ca-45	5.8E-02	5.8E+01	5.8E+04
Ca-47	1.1E-02	1.1E+01	1.1E+04
Cd-109	5.0E-03	5.0E+00	5.0E+03
Cd-113	3.3E-04	3.3E-01	3.3E+02
Cd-113m	4.4E-04	4.4E-01	4.4E+02
Cd-115	5.4E-02	5.4E+01	5.4E+04
Cd-115m	1.0E-02	1.0E+01	1.0E+04
Cd-117	5.6E-02	5.6E+01	5.6E+04
Cd-117m	1.3E-01	1.3E+02	1.3E+05
Ce-139	2.6E-03	2.6E+00	2.6E+03
Ce-141	1.8E-02	1.8E+01	1.8E+04
Ce-143	1.0E-01	1.0E+02	1.0E+05
Ce-144	1.7E-03	1.7E+00	1.7E+03
Cf-248	2.0E-05	2.0E-02	2.0E+01
Cf-249	1.7E-06	1.7E-03	1.7E+00
Cf-250	4.0E-06	4.0E-03	4.0E+00
Cf-251	1.7E-06	1.7E-03	1.7E+00
Cf-252	6.4E-06	6.4E-03	6.4E+00
Cf-253	3.3E-04	3.3E-01	3.3E+02
Cf-254	3.6E-06	3.6E-03	3.6E+00
Cl-36	1.9E-04	1.9E-01	1.9E+02
Cl-38	6.5E-01	6.5E+02	6.5E+05
Cm-242	6.0E-05	6.0E-02	6.0E+01
Cm-243	3.3E-06	3.3E-03	3.3E+00

See footnotes at the end of the table.

Table 4-1 Annual Possession Quantities for Environmental Compliance (continued)

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
Cm-244	4.2E-06	4.2E-03	4.2E+00
Cm-245	2.3E-06	2.3E-03	2.3E+00
Cm-246	2.3E-06	2.3E-03	2.3E+00
Cm-247	2.3E-06	2.3E-03	2.3E+00
Cm-248	6.4E-07	6.4E-04	6.4E-01
Cm-249	4.6E+00	4.6E+03	4.6E+06
Cm-250	1.1E-07	1.1E-04	1.1E-01
Co-56	2.4E-04	2.4E-01	2.4E+02
Co-57	1.6E-03	1.6E+00	1.6E+03
Co-58	9.0E-04	9.0E-01	9.0E+02
Co-58m	1.7E-01	1.7E+02	1.7E+05
Co-60	1.6E-05	1.6E-02	1.6E+01
Co-60m	4.0E+00	4.0E+03	4.0E+06
Co-61	3.8E+00	3.8E+03	3.8E+06
Cr-49	9.0E-01	9.0E+02	9.0E+05
Cr-51	6.3E-02	6.3E+01	6.3E+04
Cs-129	1.5E-01	1.5E+02	1.5E+05
Cs-131	2.8E-01	2.8E+02	2.8E+05
Cs-132	1.3E-02	1.3E+01	1.3E+04
Cs-134	5.2E-05	5.2E-02	5.2E+01
Cs-134m	3.2E-01	3.2E+02	3.2E+05
Cs-135	2.4E-02	2.4E+01	2.4E+04
Cs-136	2.1E-03	2.1E+00	2.1E+03
Cs-137	2.3E-05	2.3E-02	2.3E+01
Cs-138	4.4E-01	4.4E+02	4.4E+05
Cu-61	4.0E-01	4.0E+02	4.0E+05
Cu-64	5.2E-01	5.2E+02	5.2E+05
Cu-67	1.5E-01	1.5E+02	1.5E+05
Dy-157	4.4E-01	4.4E+02	4.4E+05
Dy-165	5.6E+00	5.6E+03	5.6E+06
Dy-166	8.1E-02	8.1E+01	8.1E+04
Er-169	4.0E-01	4.0E+02	4.0E+05
Er-171	3.6E-01	3.6E+02	3.6E+05
Es-253	2.6E-04	2.6E-01	2.6E+02
Es-254	2.3E-05	2.3E-02	2.3E+01
Es-254m	1.8E-03	1.8E+00	1.8E+03
Eu-152	1.6E-05	1.6E-02	1.6E+01
Eu-152m	3.5E-01	3.5E+02	3.5E+05
Eu-154	2.0E-05	2.0E-02	2.0E+01
Eu-155	5.2E-04	5.2E-01	5.2E+02

See footnotes at the end of the table.

Table 4-1 Annual Possession Quantities for Environmental Compliance (continued)

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
Eu-156	3.2E-03	3.2E+00	3.2E+03
F-18	5.6E-01	5.6E+02	5.6E+05
Fe-52	4.9E-02	4.9E+01	4.9E+04
Fe-55	1.4E-01	1.4E+02	1.4E+05
Fe-59	1.3E-03	1.3E+00	1.3E+03
Fm-254	1.8E-02	1.8E+01	1.8E+04
Fm-255	4.0E-03	4.0E+00	4.0E+03
Fr-223	1.4E-01	1.4E+02	1.4E+05
Ga-66	5.6E-02	5.6E+01	5.6E+04
Ga-67	1.1E-01	1.1E+02	1.1E+05
Ga-68	7.6E-01	7.6E+02	7.6E+05
Ga-72	3.6E-02	3.6E+01	3.6E+04
Gd-152	4.4E-06	4.4E-03	4.4E+00
Gd-153	2.0E-03	2.0E+00	2.0E+03
Gd-159	6.8E-01	6.8E+02	6.8E+05
Ge-68	2.3E-04	2.3E-01	2.3E+02
Ge-71	2.6E+00	2.6E+03	2.6E+06
Ge-77	1.0E-01	1.0E+02	1.0E+05
H-3	1.5E+01	1.5E+04	1.5E+07
Hf-181	2.5E-03	2.5E+00	2.5E+03
Hg-193m	9.5E-02	9.5E+01	9.5E+04
Hg-197	2.4E-01	2.4E+02	2.4E+05
Hg-197m	2.5E-01	2.5E+02	2.5E+05
Hg-203	5.2E-03	5.2E+00	5.2E+03
Ho-166	2.8E-01	2.8E+02	2.8E+05
Ho-166m	6.0E-06	6.0E-03	6.0E+00
I-123	4.9E-01	4.9E+02	4.9E+05
I-124	9.3E-03	9.3E+00	9.3E+03
I-125	6.2E-03	6.2E+00	6.2E+03
I-126	3.7E-03	3.7E+00	3.7E+03
I-128	9.3E+00	9.3E+03	9.3E+06
I-129	2.6E-04	2.6E-01	2.6E+02
I-130	4.6E-02	4.6E+01	4.6E+04
I-131	6.7E-03	6.7E+00	6.7E+03
I-132	2.0E-01	2.0E+02	2.0E+05
I-133	6.7E-02	6.7E+01	6.7E+04
I-134	3.2E-01	3.2E+02	3.2E+05
I-135	1.2E-01	1.2E+02	1.2E+05
In-111	4.9E-02	4.9E+01	4.9E+04
In-113m	2.1E+00	2.1E+03	2.1E+06

See footnotes at the end of the table.

Table 4-1 Annual Possession Quantities for Environmental Compliance (continued)

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
In-114m	4.9E-03	4.9E+00	4.9E+03
In-115	2.7E-04	2.7E-01	2.7E+02
In-115m	1.4E+00	1.4E+03	1.4E+06
In-116m	3.5E-01	3.5E+02	3.5E+05
In-117	1.3E+00	1.3E+03	1.3E+06
In-117m	7.6E-02	7.6E+01	7.6E+04
Ir-190	3.5E-03	3.5E+00	3.5E+03
Ir-192	9.7E-04	9.7E-01	9.7E+02
Ir-194	2.5E-01	2.5E+02	2.5E+05
Ir-194m	1.5E-04	1.5E-01	1.5E+02
K-40	6.8E-05	6.8E-02	6.8E+01
K-42	2.9E-01	2.9E+02	2.9E+05
K-43	6.0E-02	6.0E+01	6.0E+04
K-44	4.9E-01	4.9E+02	4.9E+05
Kr-79	7.0E+00	-	-
Kr-81	1.8E+02	-	-
Kr-83m	2.0E+04	-	-
Kr-85	8.4E+02	-	-
Kr-85m	1.1E+01	-	-
Kr-87	2.0E+00	-	-
Kr-88	4.2E-01	-	-
La-140	1.6E-02	1.6E+01	1.6E+04
La-141	1.1E+00	1.1E+03	1.1E+06
La-142	2.3E-01	2.3E+02	2.3E+05
Lu-177	1.4E-01	1.4E+02	1.4E+05
Lu-177m	3.5E-04	3.5E-01	3.5E+02
Mg-28	2.1E-02	2.1E+01	2.1E+04
Mn-52	3.5E-03	3.5E+00	3.5E+03
Mn-52m	5.2E-01	5.2E+02	5.2E+05
Mn-53	5.7E-02	5.7E+01	5.7E+04
Mn-54	2.5E-04	2.5E-01	2.5E+02
Mn-56	2.5E-01	2.5E+02	2.5E+05
Mo-93	1.5E-03	1.5E+00	1.5E+03
Mo-99**	5.7E-02	5.7E+01	5.7E+04
Mo-101	8.4E-01	8.4E+02	8.4E+05
Na-22	3.2E-05	3.2E-02	3.2E+01
Na-24	2.6E-02	2.6E+01	2.6E+04
Nb-90	2.5E-02	2.5E+01	2.5E+04
Nb-93m	1.2E-02	1.2E+01	1.2E+04
Nb-94	6.0E-06	6.0E-03	6.0E+00

See footnotes at the end of the table.

Table 4-1 Annual Possession Quantities for Environmental Compliance (continued)

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
Nb-95	2.3E-03	2.3E+00	2.3E+03
Nb-95m	2.0E-02	2.0E+01	2.0E+04
Nb-96	2.5E-02	2.5E+01	2.5E+04
Nb-97	1.0E+00	1.0E+03	1.0E+06
Nd-147	3.0E-02	3.0E+01	3.0E+04
Nd-149	1.1E+00	1.1E+03	1.1E+06
Ni-56	2.0E-03	2.0E+00	2.0E+03
Ni-57	2.1E-02	2.1E+01	2.1E+04
Ni-59	2.2E-02	2.2E+01	2.2E+04
Ni-63	1.4E-01	1.4E+02	1.4E+05
Ni-65	7.0E-01	7.0E+02	7.0E+05
Np-235	3.0E-02	3.0E+01	3.0E+04
Np-237	1.8E-06	1.8E-03	1.8E+00
Np-238	1.9E-02	1.9E+01	1.9E+04
Np-239	1.0E-01	1.0E+02	1.0E+05
Np-240	6.5E-01	6.5E+02	6.5E+05
Np-240m	4.7E+00	4.7E+03	4.7E+06
Os-185	9.2E-04	9.2E-01	9.2E+02
Os-191m	9.0E-01	9.0E+02	9.0E+05
Os-191	3.8E-02	3.8E+01	3.8E+04
Os-193	2.9E-01	2.9E+02	2.9E+05
P-32	1.7E-02	1.7E+01	1.7E+04
P-33	1.2E-01	1.2E+02	1.2E+05
Pa-230	6.3E-04	6.3E-01	6.3E+02
Pa-231	8.3E-07	8.3E-04	8.3E-01
Pa-233	9.3E-03	9.3E+00	9.3E+03
Pa-234	9.3E-02	9.3E+01	9.3E+04
Pb-203	8.3E-02	8.3E+01	8.3E+04
Pb-205	1.2E-02	1.2E+01	1.2E+04
Pb-209	1.1E+01	1.1E+04	1.1E+07
Pb-210	5.5E-05	5.5E-02	5.5E+01
Pb-211	1.2E-01	1.2E+02	1.2E+05
Pb-212	6.0E-03	6.0E+00	6.0E+03
Pb-214	1.2E-01	1.2E+02	1.2E+05
Pd-103	2.1E-01	2.1E+02	2.1E+05
Pd-107	8.2E-02	8.2E+01	8.2E+04
Pd-109	9.4E-01	9.4E+02	9.4E+05
Pm-143	7.6E-04	7.6E-01	7.6E+02
Pm-144	1.1E-04	1.1E-01	1.1E+02
Pm-145	5.2E-04	5.2E-01	5.2E+02

See footnotes at the end of the table.

Table 4-1 Annual Possession Quantities for Environmental Compliance (continued)

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
Pm-146	4.4E-05	4.4E-02	4.4E+01
Pm-147	2.6E-02	2.6E+01	2.6E+04
Pm-148	1.7E-02	1.7E+01	1.7E+04
Pm-148m	7.6E-04	7.6E-01	7.6E+02
Pm-149	2.8E-01	2.8E+02	2.8E+05
Pm-151	1.2E-01	1.2E+02	1.2E+05
Po-210	9.3E-05	9.3E-02	9.3E+01
Pr-142	2.8E-01	2.8E+02	2.8E+05
Pr-143	1.0E-01	1.0E+02	1.0E+05
Pr-144	1.5E+01	1.5E+04	1.5E+07
Pt-191	6.4E-02	6.4E+01	6.4E+04
Pt-193	2.1E-02	2.1E+01	2.1E+04
Pt-193m	4.8E-01	4.8E+02	4.8E+05
Pt-195m	1.4E-01	1.4E+02	1.4E+05
Pt-197	1.1E+00	1.1E+03	1.1E+06
Pt-197m	3.6E+00	3.6E+03	3.6E+06
Pu-236	7.0E-06	7.0E-03	7.0E+00
Pu-237	2.3E-02	2.3E+01	2.3E+04
Pu-238	2.7E-06	2.7E-03	2.7E+00
Pu-239	2.5E-06	2.5E-03	2.5E+00
Pu-240	2.5E-06	2.5E-03	2.5E+00
Pu-241	1.3E-04	1.3E-01	1.3E+02
Pu-242	2.5E-06	2.5E-03	2.5E+00
Pu-243	3.8E+00	3.8E+03	3.8E+06
Pu-244	2.4E-06	2.4E-03	2.4E+00
Pu-245	2.1E-01	2.1E+02	2.1E+05
Pu-246	4.8E-03	4.8E+00	4.8E+03
Ra-223	1.3E-04	1.3E-01	1.3E+02
Ra-224	3.2E-04	3.2E-01	3.2E+02
Ra-225	1.3E-04	1.3E-01	1.3E+02
Ra-226	5.5E-06	5.5E-03	5.5E+00
Ra-228	1.3E-05	1.3E-02	1.3E+01
Rb-81	4.2E-01	4.2E+02	4.2E+05
Rb-83	1.4E-03	1.4E+00	1.4E+03
Rb-84	2.0E-03	2.0E+00	2.0E+03
Rb-86	1.7E-02	1.7E+01	1.7E+04
Rb-87	1.0E-02	1.0E+01	1.0E+04
Rb-88	1.7E+00	1.7E+03	1.7E+06
Rb-89	6.4E-01	6.4E+02	6.4E+05
Re-184	1.8E-03	1.8E+00	1.8E+03

See footnotes at the end of the table.

Table 4-1 Annual Possession Quantities for Environmental Compliance (continued)

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
Re-184m	3.6E-04	3.6E-01	3.6E+02
Re-186	1.9E-01	1.9E+02	1.9E+05
Re-187	9.3E+00	9.3E+03	9.3E+06
Re-188	3.7E-01	3.7E+02	3.7E+05
Rh-103m	1.7E+02	1.7E+05	1.7E+08
Rh-105	3.4E-01	3.4E+02	3.4E+05
Ru-97	8.3E-02	8.3E+01	8.3E+04
Ru-103	3.1E-03	3.1E+00	3.1E+03
Ru-105	2.9E-01	2.9E+02	2.9E+05
Ru-106	5.9E-04	5.9E-01	5.9E+02
S-35	7.5E-02	7.5E+01	7.5E+04
Sb-117	2.0E+00	2.0E+03	2.0E+06
Sb-122	3.9E-02	3.9E+01	3.9E+04
Sb-124	6.0E-04	6.0E-01	6.0E+02
Sb-125	1.4E-04	1.4E-01	1.4E+02
Sb-126	1.8E-03	1.8E+00	1.8E+03
Sb-126m	7.6E-01	7.6E+02	7.6E+05
Sb-127	2.0E-02	2.0E+01	2.0E+04
Sb-129	1.8E-01	1.8E+02	1.8E+05
Sc-44	1.4E-01	1.4E+02	1.4E+05
Sc-46	4.0E-04	4.0E-01	4.0E+02
Sc-47	1.1E-01	1.1E+02	1.1E+05
Sc-48	1.1E-02	1.1E+01	1.1E+04
Sc-49	1.0E+01	1.0E+04	1.0E+07
Se-73	1.6E-01	1.6E+02	1.6E+05
Se-75	1.1E-03	1.1E+00	1.1E+03
Se-79	6.9E-03	6.9E+00	6.9E+03
Si-31	4.7E+00	4.7E+03	4.7E+06
Si-32	7.2E-04	7.2E-01	7.2E+02
Sm-147	1.4E-05	1.4E-02	1.4E+01
Sm-151	3.5E-02	3.5E+01	3.5E+04
Sm-153	2.4E-01	2.4E+02	2.4E+05
Sn-113	1.9E-03	1.9E+00	1.9E+03
Sn-117m	2.3E-02	2.3E+01	2.3E+04
Sn-119m	2.8E-02	2.8E+01	2.8E+04
Sn-123	1.8E-02	1.8E+01	1.8E+04
Sn-125	7.2E-03	7.2E+00	7.2E+03
Sn-126	4.7E-06	4.7E-03	4.7E+00
Sr-82	1.9E-03	1.9E+00	1.9E+03
Sr-85	1.9E-03	1.9E+00	1.9E+03

See footnotes at the end of the table.

Table 4-1 Annual Possession Quantities for Environmental Compliance (continued)

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
Sr-85m	1.5E+00	1.5E+03	1.5E+06
Sr-87m	1.2E+00	1.2E+03	1.2E+06
Sr-89	2.1E-02	2.1E+01	2.1E+04
Sr-90	5.2E-04	5.2E-01	5.2E+02
Sr-91	1.2E-01	1.2E+02	1.2E+05
Sr-92	2.5E-01	2.5E+02	2.5E+05
Ta-182	4.4E-04	4.4E-01	4.4E+02
Tb-157	2.2E-03	2.2E+00	2.2E+03
Tb-160	8.4E-04	8.4E-01	8.4E+02
Tc-95	9.0E-02	9.0E+01	9.0E+04
Tc-95m	1.4E-03	1.4E+00	1.4E+03
Tc-96	5.6E-03	5.6E+00	5.6E+03
Tc-96m	7.0E-01	7.0E+02	7.0E+05
Tc-97	1.5E-03	1.5E+00	1.5E+03
Tc-97m	7.2E-02	7.2E+01	7.2E+04
Tc-98	6.4E-06	6.4E-03	6.4E+00
Tc-99	9.0E-03	9.0E+00	9.0E+03
Tc-99m	1.4E+00	1.4E+03	1.4E+06
Tc-101	3.8E+00	3.8E+03	3.8E+06
Te-121	6.0E-03	6.0E+00	6.0E+03
Te-121m	5.3E-04	5.3E-01	5.3E+02
Te-123	1.2E-03	1.2E+00	1.2E+03
Te-123m	2.7E-03	2.7E+00	2.7E+03
Te-125m	1.5E-02	1.5E+01	1.5E+04
Te-127	2.9E+00	2.9E+03	2.9E+06
Te-127m	7.3E-03	7.3E+00	7.3E+03
Te-129	6.5E+00	6.5E+03	6.5E+06
Te-129m	6.1E-03	6.1E+00	6.1E+03
Te-131	9.4E-01	9.4E+02	9.4E+05
Te-131m	1.8E-02	1.8E+01	1.8E+04
Te-132	6.2E-03	6.2E+00	6.2E+03
Te-133	1.2E+00	1.2E+03	1.2E+06
Te-133m	2.9E-01	2.9E+02	2.9E+05
Te-134	4.4E-01	4.4E+02	4.4E+05
Th-226	3.0E-02	3.0E+01	3.0E+04
Th-227	6.4E-05	6.4E-02	6.4E+01
Th-228	2.9E-06	2.9E-03	2.9E+00
Th-229	4.9E-07	4.9E-04	4.9E-01
Th-230	3.2E-06	3.2E-03	3.2E+00
Th-231	8.4E-01	8.4E+02	8.4E+05

See footnotes at the end of the table.

Table 4-1 Annual Possession Quantities for Environmental Compliance (continued)

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
Th-232	6.0E-07	6.0E-04	6.0E-01
Th-234	2.0E-02	2.0E+01	2.0E+04
Ti-44	5.2E-06	5.2E-03	5.2E+00
Ti-45	4.0E-01	4.0E+02	4.0E+05
Tl-200	4.4E-02	4.4E+01	4.4E+04
Tl-201	1.8E-01	1.8E+02	1.8E+05
Tl-202	1.0E-02	1.0E+01	1.0E+04
Tl-204	2.5E-02	2.5E+01	2.5E+04
Tm-170	2.4E-02	2.4E+01	2.4E+04
Tm-171	5.9E-02	5.9E+01	5.9E+04
U-230	5.0E-05	5.0E-02	5.0E+01
U-231	1.4E-01	1.4E+02	1.4E+05
U-232	1.3E-06	1.3E-03	1.3E+00
U-233	7.6E-06	7.6E-03	7.6E+00
U-234	7.6E-06	7.6E-03	7.6E+00
U-235	7.0E-06	7.0E-03	7.0E+00
U-236	8.4E-06	8.4E-03	8.4E+00
U-237	4.7E-02	4.7E+01	4.7E+04
U-238	8.6E-06	8.6E-03	8.6E+00
U-239	8.3E+00	8.3E+03	8.3E+06
U-240	1.8E-01	1.8E+02	1.8E+05
V-48	1.4E-03	1.4E+00	1.4E+03
V-49	1.3E+00	1.3E+03	1.3E+06
W-181	1.1E-02	1.1E+01	1.1E+04
W-185	1.6E-01	1.6E+02	1.6E+05
W-187	1.1E-01	1.1E+02	1.1E+05
W-188	1.0E-02	1.0E+01	1.0E+04
Xe-122	7.6E-02	7.6E+01	7.6E+04
Xe-123	1.6E+00	1.6E+03	1.6E+06
Xe-125	6.0E-01	-	-
Xe-127	7.0E+00	-	-
Xe-129m	7.6E+01	-	-
Xe-131m	2.2E+02	-	-
Xe-133	5.2E+01	-	-
Xe-133m	6.0E+01	-	-
Xe-135	7.6E+00	-	-
Xe-135m	4.2E+00	-	-
Xe-138	9.9E-01	-	-
Y-86	2.8E-02	2.8E+01	2.8E+04
Y-87	2.3E-02	2.3E+01	2.3E+04

See footnotes at the end of the table.

Table 4-1 Annual Possession Quantities for Environmental Compliance (continued)

Radionuclide	Gaseous Form*	Liquid/ Powder Forms	Solid Form*
Y-88	2.5E-04	2.5E-01	2.5E+02
Y-90	1.1E-01	1.1E+02	1.1E+05
Y-90m	4.3E-01	4.3E+02	4.3E+05
Y-91	1.8E-02	1.8E+01	1.8E+04
Y-91m	1.6E+00	1.6E+03	1.6E+06
Y-92	7.0E-01	7.0E+02	7.0E+05
Y-93	3.8E-01	3.8E+02	3.8E+05
Yb-169	5.5E-03	5.5E+00	5.5E+03
Yb-175	2.1E-01	2.1E+02	2.1E+05
Zn-62	8.6E-02	8.6E+01	8.6E+04
Zn-65	4.4E-04	4.4E-01	4.4E+02
Zn-69	2.7E+01	2.7E+04	2.7E+07
Zn-69m	2.0E-01	2.0E+02	2.0E+05
Zr-86	2.4E-02	2.4E+01	2.4E+04
Zr-88	2.7E-04	2.7E-01	2.7E+02
Zr-89	1.6E-02	1.6E+01	1.6E+04
Zr-93	2.8E-03	2.8E+00	2.8E+03
Zr-95	6.4E-04	6.4E-01	6.4E+02
Zr-97	4.6E-02	4.6E+01	4.6E+04

*Radionuclides boiling at 100° C or less must be considered a gas. Capsules containing radionuclides in liquid or powder form can be considered to be solids.

**Mo-99 contained in a generator to produce Technetium-99 can be assumed to be a solid.

This procedure will demonstrate compliance with the standard if the sum of the ratios computed for each radionuclide handled is less than or equal to unity.

4.3.2 Procedure 2: Concentration Limits

The Agency has also derived a Table of Air Concentration Levels for Environmental Compliance (see Table 4-2) specifying maximum concentrations of radionuclides in the effluent air. For each radionuclide, the value given in the table is the maximum ground-level air concentration that would not result in a dose exceeding the standard. This method is extremely conservative in that it assumes that no dispersion occurs between the point of release and the point at which the most exposed individual resides, and that the most exposed individual grows all of his or her own food at that location. The table of air concentration cannot be used if the receptor is within 3 stack diameters of the point of release.

This procedure also requires very little data input by the user. The concentration in the effluent can be either measured or calculated using the EPA-approved emission factor and the volumetric flow in the vent or stack. Compliance with the standard will be demonstrated if the concentration in the stack is less than or equal to four times the value given in the table. The factor of four accounts for the fact that the wind does not always blow in the same direction; wind blowing toward the most exposed individual 25 percent of the time is a conservative upper-bound value. If more than one radionuclide is released, or if releases occur at more than one release point, the user simply calculates the ratio of the actual concentration to the concentration given in Table 4-2 for each radionuclide and/or release point and sums the results. If the sum of the ratios is four or less, compliance with the standard has been demonstrated.

Table 4-2. Concentration Levels for Environmental Compliance

Radionuclide	Concentration (Ci/m ³)	Radionuclide	Concentration (Ci/m ³)
Ac-225	9.1E-14	Bi-207	1.0E-14
Ac-227	1.6E-16	Bi-210	2.9E-13
Ac-228	3.7E-12	Bi-212	5.6E-11
Ag-106	1.9E-09	Bi-213	7.1E-11
Ag-106m	1.2E-12	Bi-214	1.4E-10
Ag-108m	7.1E-15	Bk-249	5.6E-13
Ag-110m	9.1E-14	Bk-250	9.1E-11
Ag-111	2.5E-12	Br-77	4.2E-11
Al-26	4.8E-15	Br-80	1.4E-08
Am-241	1.9E-15	Br-80m	1.8E-09
Am-242	1.5E-11	Br-82	1.2E-11
Am-242m	2.0E-15	Br-83	1.2E-08
Am-243	1.8E-15	Br-84	6.7E-10
Am-244	4.0E-11	C-11	1.5E-09
Am-245	8.3E-09	C-14	1.0E-11
Am-246	1.2E-09	Ca-41	4.2E-13
Ar-37	1.6E-03	Ca-45	1.3E-12
Ar-41	1.7E-09	Ca-47	2.4E-12
As-72	2.4E-11	Cd-109	5.9E-13
As-73	1.1E-11	Cd-113	9.1E-15
As-74	2.2E-12	Cd-113m	1.7E-14
As-76	5.0E-11	Cd-115	1.6E-11
As-77	1.6E-10	Cd-115m	8.3E-13
At-211	1.1E-11	Cd-117	6.7E-11
Au-193	3.8E-10	Cd-117m	1.6E-10
Au-194	3.2E-11	Ce-139	2.6E-12
Au-195	3.1E-12	Ce-141	6.3E-12
Au-198	2.1E-11	Ce-143	3.0E-11
Au-199	4.8E-11	Ce-144	6.2E-13
Ba-131	7.1E-12	Cf-248	1.8E-14
Ba-133	5.9E-14	Cf-249	1.4E-15
Ba-133m	5.9E-11	Cf-250	3.2E-15
Ba-135m	1.8E-10	Cf-251	1.4E-15
Ba-139	5.6E-09	Cf-252	5.6E-15
Ba-140	1.3E-12	Cf-253	3.1E-13
Ba-141	1.4E-09	Cf-254	3.0E-15
Ba-142	1.3E-09	Cl-36	2.7E-15
Be-7	2.3E-11	Cl-38	7.7E-10
Be-10	1.6E-12	Cm-242	5.3E-14
Bi-206	2.3E-12	Cm-243	2.6E-15

Table 4-2. Concentration Levels for Environmental Compliance (continued)

Radionuclide	Concentration (Ci/m ³)	Radionuclide	Concentration (Ci/m ³)
Cm-244	3.3E-15	Eu-156	1.9E-12
Cm-245	1.8E-15	F-18	6.7E-10
Cm-246	1.9E-15	Fe-52	5.6E-11
Cm-247	1.9E-15	Fe-55	9.1E-12
Cm-248	5.0E-16	Fe-59	6.7E-13
Cm-249	3.7E-09	Fm-254	2.0E-11
Cm-250	9.1E-17	Fm-255	4.3E-12
Co-56	1.8E-13	Fr-223	3.3E-11
Co-57	1.3E-12	Ga-66	6.2E-11
Co-58	6.7E-13	Ga-67	7.1E-11
Co-58m	1.2E-10	Ga-68	9.1E-10
Co-60	1.7E-14	Ga-72	3.8E-11
Co-60m	4.3E-09	Gd-152	5.0E-15
Co-61	4.5E-09	Gd-153	2.1E-12
Cr-49	1.1E-09	Gd-159	2.9E-10
Cr-51	3.1E-11	Ge-68	2.0E-13
Cs-129	1.4E-10	Ge-71	2.4E-10
Cs-131	3.3E-11	Ge-77	1.0E-10
Cs-132	4.8E-12	H-3	1.5E-09
Cs-134	2.7E-14	Hf-181	1.9E-12
Cs-134m	1.7E-10	Hg-193m	1.0E-10
Cs-135	4.0E-13	Hg-197	8.3E-11
Cs-136	5.3E-13	Hg-197m	1.1E-10
Cs-137	1.9E-14	Hg-203	1.0E-12
Cs-138	5.3E-10	Ho-166	7.1E-11
Cu-61	4.8E-10	Ho-166m	7.1E-15
Cu-64	5.3E-10	I-123	4.3E-10
Cu-67	5.0E-11	I-124	6.2E-13
Dy-157	5.0E-10	I-125	1.2E-13
Dy-165	6.7E-09	I-126	1.1E-13
Dy-166	1.1E-11	I-128	1.1E-08
Er-169	2.9E-11	I-129	9.1E-15
Er-171	4.0E-10	I-130	4.5E-11
Es-253	2.4E-13	I-131	2.1E-13
Es-254	2.0E-14	I-132	2.3E-10
Es-254m	1.8E-12	I-133	2.0E-11
Eu-152	2.0E-14	I-134	3.8E-10
Eu-152m	3.6E-10	I-135	1.2E-10
Eu-154	2.3E-14	In-111	3.6E-11
Eu-155	5.9E-13	In-113m	2.5E-09

Table 4-2. Concentration Levels for Environmental Compliance (continued)

Radionuclide	Concentration (Ci/m ³)	Radionuclide	Concentration (Ci/m ³)
In-114m	9.1E-13	Nb-95	2.2E-12
In-115	7.1E-14	Nb-95m	1.4E-11
In-115m	1.6E-09	Nb-96	2.4E-11
In-116m	4.2E-10	Nb-97	1.2E-09
In-117	1.6E-09	Nd-147	7.7E-12
In-117m	9.1E-11	Nd-149	7.1E-10
Ir-190	2.6E-12	Ni-56	1.7E-12
Ir-192	9.1E-13	Ni-57	1.8E-11
Ir-194	1.1E-10	Ni-59	1.5E-11
Ir-194m	1.7E-13	Ni-63	1.4E-11
K-40	2.7E-14	Ni-65	8.3E-10
K-42	2.6E-10	Np-235	2.5E-11
K-43	6.2E-11	Np-237	1.2E-15
K-44	5.9E-10	Np-238	1.4E-11
Kr-79	8.3E-09	Np-239	3.8E-11
Kr-81	2.1E-07	Np-240	7.7E-10
Kr-83m	2.3E-05	Np-240m	5.6E-09
Kr-85	1.0E-06	Os-185	1.0E-12
Kr-85m	1.3E-08	Os-191m	2.9E-10
Kr-87	2.4E-09	Os-191	1.1E-11
Kr-88	5.0E-10	Os-193	9.1E-11
La-140	1.2E-11	P-32	3.3E-13
La-141	7.7E-10	P-33	2.4E-12
La-142	2.7E-10	Pa-230	3.2E-13
Lu-177	2.4E-11	Pa-231	5.9E-16
Lu-177m	3.6E-13	Pa-233	4.8E-12
Mg-28	1.5E-11	Pa-234	1.1E-10
Mn-52	2.8E-12	Pb-203	6.2E-11
Mn-52m	6.2E-10	Pb-205	5.6E-12
Mn-53	1.5E-11	Pb-209	1.3E-08
Mn-54	2.8E-13	Pb-210	2.8E-15
Mn-56	2.9E-10	Pb-211	1.4E-10
Mo-93	1.1E-12	Pb-212	6.3E-12
Mo-99	1.4E-11	Pb-214	1.2E-10
Mo-101	1.0E-09	Pd-103	3.8E-11
Na-22	2.6E-14	Pd-107	3.1E-11
Na-24	2.6E-11	Pd-109	4.8E-10
Nb-90	2.6E-11	Pm-143	9.1E-13
Nb-93m	1.0E-11	Pm-144	1.3E-13
Nb-94	7.1E-15	Pm-145	6.2E-13

Table 4-2. Concentration Levels for Environmental Compliance (continued)

Radionuclide	Concentration (Ci/m ³)	Radionuclide	Concentration (Ci/m ³)
Pm-146	5.3E-14	Re-184m	3.7E-13
Pm-147	1.1E-11	Re-186	1.8E-11
Pm-148	5.0E-12	Re-187	2.6E-10
Pm-148m	6.7E-13	Re-188	1.7E-10
Pm-149	4.2E-11	Rh-103m	2.1E-07
Pm-151	7.1E-11	Rh-105	1.3E-10
Po-210	7.1E-15	Ru-97	6.7E-11
Pr-142	1.1E-10	Ru-103	2.6E-12
Pr-143	7.1E-12	Ru-105	2.8E-10
Pr-144	1.8E-08	Ru-106	3.4E-13
Pt-191	4.3E-11	S-35	1.3E-12
Pt-193	1.8E-11	Sb-117	2.4E-09
Pt-193m	4.8E-11	Sb-122	1.4E-11
Pt-195m	3.2E-11	Sb-124	5.3E-13
Pt-197	4.0E-10	Sb-125	1.6E-13
Pt-197m	2.6E-09	Sb-126	1.4E-12
Pu-236	5.9E-15	Sb-126m	9.1E-10
Pu-237	1.9E-11	Sb-127	7.1E-12
Pu-238	2.1E-15	Sb-129	7.7E-11
Pu-239	2.0E-15	Sc-44	1.7E-10
Pu-240	2.0E-15	Sc-46	4.2E-13
Pu-241	1.0E-13	Sc-47	3.8E-11
Pu-242	2.0E-15	Sc-48	9.1E-12
Pu-243	4.2E-09	Sc-49	1.2E-08
Pu-244	2.0E-15	Se-73	1.7E-10
Pu-245	2.1E-10	Se-75	1.7E-13
Pu-246	2.2E-12	Se-79	1.1E-13
Ra-223	4.2E-14	Si-31	5.6E-09
Ra-224	1.5E-13	Si-32	3.4E-14
Ra-225	5.0E-14	Sm-147	1.4E-14
Ra-226	3.3E-15	Sm-151	2.1E-11
Ra-228	5.9E-15	Sm-153	5.9E-11
Rb-81	5.0E-10	Sn-113	1.4E-12
Rb-83	3.4E-13	Sn-117m	5.6E-12
Rb-84	3.6E-13	Sn-119m	5.3E-12
Rb-86	5.6E-13	Sn-123	1.1E-12
Rb-87	1.6E-13	Sn-125	1.7E-12
Rb-88	2.1E-09	Sn-126	5.3E-15
Rb-89	7.1E-10	Sr-82	6.2E-13
Re-184	1.5E-12	Sr-85	1.8E-12

Table 4-2. Concentration Levels for Environmental Compliance (continued)

Radionuclide	Concentration (Ci/m ³)	Radionuclide	Concentration (Ci/m ³)
Sr-85m	1.6E-09	Th-232	6.2E-16
Sr-87m	1.4E-09	Th-234	2.2E-12
Sr-89	1.8E-12	Ti-44	6.2E-15
Sr-90	1.9E-14	Ti-45	4.8E-10
Sr-91	9.1E-11	Tl-200	4.5E-11
Sr-92	2.9E-10	Tl-201	1.0E-10
Ta-182	4.5E-13	Tl-202	5.0E-12
Tb-157	2.5E-12	Tl-204	1.2E-12
Tb-160	7.7E-13	Tm-170	3.3E-12
Tc-95	1.0E-10	Tm-171	2.6E-11
Tc-95m	1.4E-12	U-230	1.5E-14
Tc-96	5.6E-12	U-231	4.2E-11
Tc-96m	6.7E-10	U-232	1.3E-15
Tc-97	7.1E-13	U-233	7.1E-15
Tc-97m	7.1E-12	U-234	7.7E-15
Tc-98	6.7E-15	U-235	7.1E-15
Tc-99	1.4E-13	U-236	7.7E-15
Tc-99m	1.7E-09	U-237	1.0E-11
Tc-101	4.5E-09	U-238	8.3E-15
Te-121	1.0E-12	U-239	4.3E-09
Te-121m	1.2E-13	U-240	1.3E-10
Te-123	1.4E-13	V-48	1.0E-12
Te-123m	2.0E-13	V-49	1.6E-10
Te-125m	3.6E-13	W-181	6.7E-12
Te-127	1.0E-09	W-185	2.6E-12
Te-127m	1.5E-13	W-187	7.7E-11
Te-129	7.7E-09	W-188	5.3E-13
Te-129m	1.4E-13	Xe-122	9.1E-11
Te-131	9.1E-11	Xe-123	1.6E-09
Te-131m	1.0E-12	Xe-125	1.1E-11
Te-132	7.1E-13	Xe-127	8.3E-09
Te-133	9.1E-10	Xe-129m	9.1E-08
Te-133m	2.2E-10	Xe-131m	2.6E-07
Te-134	5.3E-10	Xe-133	6.2E-08
Th-226	3.4E-11	Xe-133m	7.1E-08
Th-227	3.8E-14	Xe-135	9.1E-09
Th-228	3.1E-15	Xe-135m	5.0E-09
Th-229	5.3E-16	Xe-138	1.2E-09
Th-230	3.4E-15	Y-86	3.0E-11
Th-231	2.9E-10	Y-87	1.7E-11

Table 4-2. Concentration Levels for Environmental Compliance (continued)

Radionuclide	Concentration (Ci/m ³)	Radionuclide	Concentration (Ci/m ³)
Y-88	2.7E-13	Zn-65	9.1E-14
Y-90	1.3E-11	Zn-69	3.2E-08
Y-90m	1.9E-10	Zn-69m	1.7E-10
Y-91	2.1E-12	Zr-86	2.4E-11
Y-91m	1.3E-09	Zr-88	3.1E-13
Y-92	8.3E-10	Zr-89	1.3E-11
Y-93	2.9E-10	Zr-93	2.6E-12
Yb-169	3.7E-12	Zr-95	6.7E-13
Yb-175	4.3E-11	Zr-97	3.8E-11
Zn-62	9.1E-11		

This procedure also requires very little data input by the user. The concentration in the effluent can be either measured or calculated using the EPA-approved emission factor and the volumetric flow in the vent or stack. Compliance with the standard will be demonstrated if the concentration in the stack is less than or equal to four times the value given in the table. The factor of four accounts for the fact that the wind does not always blow in the same direction; wind blowing toward the most exposed individual 25 percent of the time is a conservative upper-bound value. If more than one radionuclide is released, or if releases occur at more than one release point, the user simply calculates the ratio of the actual concentration to the concentration given in Table 4-2 for each radionuclide and/or release point and sums the results. If the sum of the ratios is four or less, compliance with the standard has been demonstrated.

4.3.3 Procedure 3: NCRP Screening Procedures

The third procedure is to calculate doses using the screening procedures developed by the NCRP. The "Screening Levels" presented in Commentary No. 3 all begin with the quantities of radionuclides released to the air from the facility. These quantities may be determined using any EPA-approved method.

The three procedures follow a tiered approach, in which the conservative assumptions regarding the dispersion of the effluent and the location of the farm producing foodstuffs are relaxed as the user supplies additional site-specific information. At the first level, the user simply provides the release rates for each nuclide and the volumetric flow of the release point. The concentration of radionuclides at the point of release is used to determine the exposure for all pathways. The effective whole-body dose equivalent is computed by applying the radionuclide-specific

screening factors supplied by the NCRP to the estimated concentrations and summing the results. These screening factors incorporate the terrestrial transport models and dosimetric models described in Section 4.2 and include doses from all pathways. If compliance is demonstrated at this level, the user stops.

If compliance is not shown, the user moves to the second level. At this level, the user, guided by a decision diagram, selects the appropriate air dispersion model for the facility and calculates the concentration of radionuclides in air at the distance where a receptor lives. The manner in which the calculation is done ensures that the assessment is made at the point of maximum concentration. The same screening factors used in the first level are then applied to this concentration to compute the resulting effective whole-body dose equivalents. Again, if compliance is not demonstrated, the user goes on to the third level.

At the third level, the user identifies the location of the nearest farms producing foodstuffs and calculates the concentrations of radionuclides in air for those locations using the appropriate dispersion model. Screening factors that consider only the ingestion pathway are then applied to this concentration to determine the dose from the ingestion pathway. Screening factors that include only air immersion, inhalation, and ground-surface contamination are applied to the concentration in air at the location where the most exposed individual resides. The doses from all pathways are then summed to determine compliance.

4.3.4 Procedure 4: Compliance Model of the COMPLY Computer Code

The Agency has developed the COMPLY computer code to assist the regulated community in determining compliance with the limits of the NESHAP. The COMPLY code is a computer implementation of the first three compliance procedures and includes an additional

level - the "Compliance Model." The "Compliance Model," which is an extension of the NCRP's procedures, differs from the third level of the NCRP's Screening Procedures in several ways. These differences are documented fully in EPA89b.

The air dispersion models account for plume rise and allow the user to input site-specific meteorological data in the form of a wind rose showing actual frequencies of wind in each of the 16 directions and the actual mean wind speed for each direction. Doses resulting from the ingestion pathway are more realistically calculated by allowing the user to specify the location of the nearest farm producing vegetables and the nearest farm producing beef and milk. Finally, more realistic environmental parameters (AIRDOS-EPA default values) are used.

The COMPLY code is easy to run, guiding the user with on-screen messages and prompts for the needed data. The user may begin at the first level and proceed to higher levels as needed to demonstrate compliance. For each assessment mode, the code compares the computed doses with the limits of the standard to determine whether compliance has been demonstrated. The code also supplies the user with a hard copy report showing the input values supplied and a summary of the computed doses. The code, written in FORTRAN, can be run on any IBM-PC or PC-compatible computer running MS-DOS version 2.0 or later, and having at least 512 kilobytes of memory and either one floppy disk drive and a hard disk or two floppy disk drives.

4.4 EXEMPTION CRITERIA

Facilities covered by the NESHAP are subject to the reporting and approval requirements of Part 61, Subpart I, Sections 61.104(a) and 61.106(a). The Agency has determined that these requirements could represent an unnecessary burden on small users. Therefore,

the Agency has adopted the following exemption criteria for facilities covered by the these requirements:

- o An existing facility will be exempt if the effective dose equivalent that is caused by all emissions from the facility is less than 1 mrem per year.
- o Any new construction or modification of an existing facility will not need to file an application for approval if one of the following conditions is met:
 1. The effective dose equivalent that is caused by all emissions from the facility is less than 1 mrem per year.
 2. The effective dose equivalent that is caused by all emissions from the new construction or modification is less than 0.1 mrem per year.

Exemption may be determined using the Table of Annual Possession Quantities for Environmental Compliance, the Table of Air Concentration for Environmental Compliance, Screening Levels 1-3 of NCRP Commentary No. 3, or the COMPLY Code.

Exemption may not be determined using other EPA-approved procedures. The EPA believes it is reasonable to require facilities using other methods to submit their input data and results for review.

REFERENCES

- Ba76 Baker, D.A., Hoenes, G.R., and Soldat, J.R., "FOOD--An Interactive Code to Calculate Internal Radiation Doses from Contaminated Food Products," Proc. of Conference on Environmental Modeling and Simulation, April 19-22, 1976, Cincinnati, Ohio, EPA 600/9-76-016, July 1976.
- BR74 Briggs, G.A., "Diffusion Estimation for Small Emissions," Environmental Research Laboratories, Air Resources Atmospheric Turbulence and Laboratory 1973 Annual Report, U.S. Atomic Energy Commission Report ATDL-106, National Oceanic and Atmospheric Administration, Oak Ridge, TN, 1974.
- DOE88 U.S. Department of Energy, "External Dose Conversion Factors for Calculation of Dose to the Public," DOE/EH-0070, July 1988.
- EPA88 "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Immersion, and Ingestion," Federal Guidance Report No. 11, EPA 520/1-88-020, September 1988.
- EPA89 U.S. Environmental Protection Agency, Risk Assessments: Environmental Impact Statement for NESHAPS - Radio-nuclides, Volume 1, Office of Radiation Programs, Washington, DC, September 1989.
- EPA89a U.S. Environmental Protection Agency, EPA Guidance Document for Facilities Subject to 40 CFR Part 61, Subpart I: Procedures for Determining Compliance with the Standard and Qualification for Exemption From Reporting,

- EPA 520/1-89-002, Office of Radiation Programs, Washington DC, October 1989.
- EPA89b U.S. Environmental Protection Agency, User's Guide for COMPLY, EPA 520/1-89-003, Office of Radiation Programs, Washington DC, October 1989.
- FI80 Fields, D.E., and Miller, C.W., User's Manual for DWNWND - An Interactive Gaussian Plume Atmospheric Transport Model with Dispersion Parameter Options, DOE Report ORNL/TM-6874, Oak Ridge National Laboratory, Oak Ridge, TN, 1980.
- GI68 Gifford, F.A., Jr., "An Outline of Theories of Diffusion in the Lower Layers of the Atmosphere," Meteorology and Atomic Energy - 1968, U.S. Atomic Energy Commission Report TID-2419, Slade, D., Ed., U.S. Atomic Energy Commission, Washington DC, 1968.
- GI75 Gifford, F.A., Jr., "Turbulent Diffusion Typing Schemes - A Review", Nuclear Safety, Vol. 17, No. 68, 1975.
- IA82 International Atomic Energy Agency, Generic Models and Parameters for Assessing the Environmental Transfer of Radionuclides from Routine Releases, Safety Series No. 57, Vienna, Austria, 1982.
- MI84 Miller, C.W., and Yildiran, M., "Estimating Radionuclide Air Concentrations Near Buildings: A Screening Approach," Transactions of the American Nuclear Society, Vol. 46, No. 55, 1984.
- NCRP84 The National Council on Radiation Protection and Measurements, Radiological Assessment: Predicting the Transport,

Bioaccumulation, and Uptake by Man of Radionuclides Released to the Environment, NCRP Report No. 76, Bethesda, MD, March 15, 1984.

NCRP89 The National Council on Radiation Protection and Measurements, Screening Techniques for Determining Compliance with Environmental Standards: Releases of Radionuclides to the Atmosphere, NCRP Commentary No. 3, Revision of January 1989, Bethesda, MD, January 1989.

Sk74, Skrabble, K.W., et al., "A General Equation for the Kinetics of Linear First Order Phenomena and Suggested Applications," Health Physics, 27, 155, 1974.

WI82 Wilson, D.J., and Britter, R.E., "Estimates of Building Surface Concentrations from Nearby Point Sources," Atmos. Environ., Vol. 16, No. 2631, 1982.

APPENDIX A: ADDITIONAL REFERENCES

The following references, in addition to those listed in the text, were reviewed and used in developing the material presented in this report.

Alexander, R.E., Neel, R.B., Puskin, J.S., and Brodsky, A., "Internal Dosimetry Model for Applications to Bioassay at Uranium Mills," NUREG-0874, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 1986.

Amersham Corporation, "Products and Services for the Life Sciences," Arlington Heights, IL, 1986.

Barnes, D.E., "Basic Criteria in the Control of Air and Surface Contamination," in Health Physics in Nuclear Installations (Proc. Symp. Ris, 1959), OECD/ENEA, Paris, 1959.

Bennett, D.E., Runkle, G.E., Alpert, D.J., Johnson, J.D., and Harlan, C.P., "Preliminary Screening of Fuel Cycle and By-Product Material Licenses for Emergency Planning," NUREG/CR-3657, Sandia National Laboratories, Albuquerque, NM 87185 and Livermore, CA 94550, March 1985, pp. 10-13, pp. 22-25.

Bremer, P.O., "Pharmaceutical Form--Packaging," Chapter 4 in Safety and Efficiency of Radiopharmaceuticals, Martinus Nijhoff Publishers, Boston, 1984.

Brodsky, A., "Information for Establishing Bioassay Measurements and Evaluations of Tritium Exposure," NUREG-0938, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 1983.

Brodsky, A., "Principles and Practices for Keeping Occupational Radiation Exposures at Medical Institutions As Low As Reasonably

Achievable," NUREG-0267, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 1982.

Brodsky, A., "Models for Calculating Doses from Radioactive Materials Released to the Environment," Section 6.4 in "Handbook of Radiation Measurement and Protection, Vol.II, Section A - Biological and Mathematical Information," Edited by A. Brodsky, CRC Press, Boca Raton, FL, 1982, pp. 367-422.

Brodsky, A., "Resuspension Factors and Probabilities of Intake of Material in Process (or, 'Is $E(-6)$ a Magic Number in Health Physics?')," Health Physics, 39, 992-1000, 1980.

Brodsky, A., "Experience with Intakes of Tritium from Various Processes," Health Physics, 33, 94-98, 1977.

Brodsky, A., "Determining Industrial Hygiene Requirements in Installations Using Radioactive Materials," in Handbook of Laboratory Safety, N.V. Steere, Editor, CRC Press, Boca Raton, FL, 1970, pp. 482-502.

Brodsky, A., "Determination of Facilities, Equipment and Procedures Required for Various Types of Operations," in Handbook of Radioactive Nuclides, Y. Wang, Editor, CRC Press, Boca Raton, FL, 1969, pp. 677-679.

Brodsky, A., "Determining Industrial Hygiene Requirements for Installations Using Radioactive Materials," Am. Indust. Hygiene Assn. Journ, 26, 294-310, May-June, 1965.

Brodsky, A., and Beard, G.V., Editors, "A Compendium of Information for Use in Controlling Radiation Emergencies," TID-8206(Rev.), U.S. Atomic Energy Commission, Washington, DC, 1960, 100 pp.

Brodsky, A., Sayeg, J.A., Wald, N., Wechsler R. and Caldwell, R., "The Measurement and Management of Insoluble Plutonium-Ameridium Inhalation in Man," in Proceedings of the First International Congress of Radiation Protection, W.S. Snyder et al., Editors.

Brodsky, A., Wald, N., Horm, I.S., and Varzaly, B.J., "The Removal of Am-241 from Humans by DTPA," (abstract) Health Physics, 379, 1969; also in detail in Dept. Radiation Health, GSPH, University of Pittsburgh, Pittsburgh, PA 15261, report on contract RH 00545-02, PHS, by A. Brodsky and I. Horm, 1968-71.

Brunskill, R.T., "Relationship Between Surface and Airborne Contamination," Surface Contamination (Proc. Symp. Gatlinburg, Tenn.), Pergamon Press, Oxford, 1967.

Burchsted, C.A., Kahn, J.E., and Fuller, A.B., "Nuclear Air Cleaning Handbook," ERDA 76-21, Contract No. W-7405-ENG-26, Oak Ridge National Laboratory, Oak Ridge, TN 37830, 1976.

Cember, H., Introduction to Health Physics, 2nd Edition, Pergamon Press, New York, 1983, pp. 330-331.

Committee on Industrial Ventilation, ACGIH, "Industrial Ventilation, 19th Edition, A Manual of Recommended Practice," American Conference of Governmental Industrial Hygienists, 6500 Glenway, Bldg. D-7, Cincinnati, OH 45211, 1986.

Cole, L.W. et al., "Environmental Survey of the Mallinckrodt Diagnostics Facility, Maryland Heights, Missouri," prepared for Division of Fuel and Material Safety, U.S. Nuclear Regulatory Commission, by the Radiological Site Assessment Program, Manpower Education, Research, and Training Division, Oak Ridge Associated Universities, Oak Ridge, TN 37830, March 1982, pp. 8-9, 23-26.

Dunster, H. J., "The Concept of Derived Working Limits for Surface Contamination," in B.R. Fish, pp. 139-147.

Fasiska, B., "Radiation Safety Procedures and Contamination Control Practices Involved in High Level I-131 Thyroid Therapy Cases," in P.L. Carson et al.

Fish, B.R., Editor, "Surface Contamination, Proc. Int. Symp. on Surface Contamination, Gatlinburg, Tennessee in June 1964," Pergamon Press, NY, 1967.

U.S. Food and Drug Administration, "Abbreviated Summary of Approved Radiopharmaceutical Drug Products," 1986.

Frame, P.W., "Fume Hood Design and Testing," continuing education lecture presented at 1987 Health Physics Society meeting, available from Oak Ridge Associated Universities, Oak Ridge, TN 37830, July 6, 1987.

Franke, T., and Hunzinger, W., "Statistical Investigation into Amounts of Radionuclides Accidentally Inhaled," in Diagnosis and Treatment of Deposited Radionuclides, Edited by H. A. Kornberg and W. D. Norwood, Excerpta Medica Foundation, Amsterdam, 1968, pp. 457-459.

Frost, D. and Jammet, H., Manual on Radiation Protection in Hospitals and General Practice, Volume 2, "Unsealed Sources," World Health Organization, 1975.

Fukuda, S., Naritomi, M., Izawa, S., and Izumi, Y., "Airborne Iodine Monitoring at the Radioisotope Test Production Plant, JAERI," in W.S. Snyder, et al., pp. 1153-1166.

Gallagher, B., Ph.D., New England Nuclear, personal communication, June 1986.

U.S. Department of Health and Human Services, Workshop Manual for Radionuclide Handling and Radiopharmaceutical Quality Assurance, 1983.

Healy, J.W., "Surface Contamination: Decision Levels," LA-4558-MS, Los Alamos Scientific Laboratory, Los Alamos, NM 87544, 1971, pp. 30-34 and Appendix C.

Heid, K., Chairman, Working Group 2.5, Health Physics Society Standards Committee, "Performance Criteria for Radiobioassay," Draft Health Physics Society Standard and draft ANSI N13.30 Standard, available from Health Physics Society, 1340 Old Chain Bridge Road, McLean, VA 22101, 1987.

Holcomb, R.J., et al., "Radiation Safety Program at the National Institutes of Health," Nuclear Safety, Volume 25, No. 5, pp. 676-688, 1984.

Howard, B.Y., "Safe Handling of Radioiodine Solutions," in Operational Health Physics, Proceedings of the Ninth Midyear Topical Symposium of the Health Physics Society, Denver, CO, Feb. 1976, Edited by P.L. Carson, W.R. Hendee, and D.C. Hunt, Central Rocky Mountain Chapter, Health Physics Society, P. O. Box 3229, Boulder, CO 80303, 1976, pp. 247-249.

Howell, W.P., "Radiation Protection Aspects of Work with Po-210," in C.A. Willis and J.S. Handloser, pp. 539-562.

Hupf, H.B., "Radiopharmaceuticals for Clinical Use," Chapter 2 in Practical Nuclear Medicine, Fuad S. Asnkar, Editor, Medcom Medical Update Series, Medcom Press, 1974.

ICN Biomedicals, Inc., ICN Radiochemicals, Irvine, CA, no date.

International Atomic Energy Agency (IAEA), "Radiological Surveillance of Airborne Contaminants in the Working Environment," IAEA Safety Series No. 49, Procedures and Data, IAEA, Vienna, Austria, 1979.

International Atomic Energy Agency, "Monitoring of Radioactive Contamination on Surfaces," Technical Report Series No. 120, 1970.

International Atomic Energy Agency, "Safe Handling of Radioisotopes," Safety Series No. 1, International Atomic Energy Agency, Vienna, Austria, 1958, p. 99, pp. 35-36.

International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Supplement to Part 1, Pergamon Press, Oxford, 1980.

International Commission on Radiological Protection, "Limits for Intakes of Radionuclides by Workers," ICRP Publication 30, Part 1, Pergamon Press, Oxford, 1979.

International Commission on Radiological Protection, "Recommendations of the International Commission on Radiological Protection," ICRP Publication 26, Pergamon Press, Oxford, 1977.

International Commission on Radiological Protection, "The Handling, Storage, Use and Disposal of Unsealed Radionuclides in Hospitals and Medical Research Establishments," ICRP Publication 25, Pergamon Press, Oxford, 1976.

International Commission on Radiological Protection, "Report of the ICRP Task Group on Lung Dynamics," Health Physics, 12, 173-207, 1966.

International Commission on Radiological Protection, "Report of Committee II on Permissible Dose for Internal Radiation," ICRP Publication 2 (sometimes abbreviated ICRP-2), Health Physics, 3, 1-380, June 1960 (also available from Pergamon Press, NY).

Johnson, A.S., "Autopsy Experience with a Radioactive Cadaver," Health Physics, 37, pp. 231-236, 1979.

Jones, I.S., and Pond, S.F., "Some Experiments to Determine the Resuspension Factor of Plutonium from Various Surfaces," in B.R. Fish, pp. 83-92.

Jones, I.S., Pond, S.F., and Stevens, D.C., "Resuspension Factors for Plutonium," in Proc. Into. Symp. Radiological Protection of the Worker by the Design and Control of His Environment, Bournemouth, Paper No. 8, 1966.

Knapp, F.F., and Butler, T.A., Editors, "Radionuclide Generators: New Systems for Nuclear Medicine Applications," ACS Symposium Series 241, American Chemical Society, Washington, DC, 1984.

Leventhal, L., et al., "Assessment of Radiopharmaceutical Usage and Release Practices by Eleven Western Hospitals," in Effluent and Environmental Radiation Surveillance, STP 698, American Society for Testing and Materials, Philadelphia, Pennsylvania, pp. 5-19, 1980.

Mallinckrodt, Inc., "Product and Physical Data--Radiopharmaceuticals," St. Louis, MO., 1985.

McGuire, S., "A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees," NUREG-1140

(Draft), U.S. Nuclear Regulatory Commission, Washington, DC 20555, Reprinted March 1986.

Medi-Physics, "Technical Product Descriptions," Richmond, CA, 1986.

Medical Economics Company, Litton Division, Physician's Desk Reference for Radiology and Nuclear Medicine 1978-9, Oradell, NJ, 1978.

Merck Company, Merck Index, 1983.

Moss, C.E., "Control of Radioisotope Release to the Environment from Diagnostic Isotope Procedures," Health Physics, 25, pp. 197-198, 1973.

Mishima, J., "A Review of Research on Plutonium Releases during Overheating and Fires," HW-83668, Hanford Laboratories, Richland, WA, Aug. 1964.

New England Nuclear, "Radiopharmaceuticals and Nuclear Medicine Sources," North Billerica, MA, undated.

New England Nuclear, "Research Products 1985-6," North Billerica, MA, 1985.

New England Nuclear, "Sources and Accessories for Nuclear Medicine," North Billerica, MA, 1983.

U.S. Nuclear Regulatory Commission, "Bioassay at Uranium Mills," Regulatory Guide 8.22, Proposed Revision 1, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555, 1987.

U.S. Nuclear Regulatory Commission, 10 CFR Parts 19 et al., "Standards for Protection Against Radiation; Proposed Rule; Extension of Comment Period and Republication," Federal Register Vol. 51, No. 6, Thursday, January 9, 1986, pp. 1142-43.

U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, Energy, Part 20, "Standards for Protection Against Radiation," Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402, 1986.

U.S. Nuclear Regulatory Commission, "Guide for the Preparation of Applications for Nuclear Pharmacy Licenses," Draft Regulatory Guide and Value/Impact Statement, For Comment, Division 10, Task FC 410-4, Director, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555, August 1985, pp. 16-17.

U.S. Nuclear Regulatory Commission, "Health Physics Surveys at Medical Institutions," Regulatory Guide 8.23, Rev. 1, 1981.

U.S. Nuclear Regulatory Commission, "Guide for Preparation of License Applications for Medical Programs," Regulatory Guide 10.8, Revision 1, 1980.

U.S. Nuclear Regulatory Commission, "Applications for Type A Licenses of Broad Scope," Regulatory Guide 10.5, Revision 1, Dec. 1980, p. 10.5-4.

U.S. Nuclear Regulatory Commission, "Health Physics Surveys for Byproduct Material at NRC-Licensed Processing and Manufacturing Plants," Regulatory Guide 8.21, Revision 1, 1979.

U.S. Nuclear Regulatory Commission, Radioactivity in Consumer Products, NUREG/CP-0001, Washington, DC, 1978.

U.S. Nuclear Regulatory Commission, "Standard Format and Content of License Applications for Plutonium Processing and Fuel Fabrication Plants," Regulatory Guide 3.39, NUREG-0010, Jan. 1976, pp. 4-9, 4-10, 5-4, 7-1.

Nyshiyama, H. et al., "Survey of Tc-99m Contamination of Laboratory Personnel: Its Degree and Routes," Radiology, 135, pp. 467-471, 1980.

Pinson, E.A., and Langham, W.H., "Physiology and Toxicology of Tritium in Man," Journal of Applied Physiology: Respiratory, Environmental and Exercise Physiology, 108-126, 1967.

Rayudu, G., editor, Radiotracers for Medical Applications: Volumes 1 and 2, CRC Press, Inc., Boca Raton, FL, 1983.

Reba, Richard C., M.D., Director of George Washington University Nuclear Medicine Division, personal communication, June 1986.

Ritter, P.D., Alvarez, J.L., and Novick, V.J., "Interpretation of Work Area and Environmental Sampling," in Proceedings of the Department of Energy Workshop on Workplace Aerosol Monitoring, Oct. 28-30, 1985, PNL-SA-14225, Edited by J. L. Kenoyer, E. J. Vallario and B. L. Murphy, Pacific Northwest Laboratory, Richland, WA 99352, Feb. 1987.

Rosen, J.C. et al., "Long-term Removal of Am-241 Using Ca-DTPA," Health Physics, 601-609, 1980.

Sansone, E., "Redispersion of Indoor Surface Contamination and Its Implications," chapter in press (Dr. Eric B. Sansone, Environmental Control and Research Program, NCI-Frederick Cancer Research Facility, Program Resources, Inc., Frederick, MD 21701, 1987.

Snyder, W.S., Chairman, "Task Group Report on Reference Man," ICRP Publication 23, International Commission on Radiological Protection, Pergamon Press, Oxford, 1975.

Stewart, N., "The Resuspension of Particulate Materials from Surfaces," in B.R. Fish, pp. 63-74.

Squibb Diagnostics, Technical Product Descriptions, New Brunswick, NJ, 1985.

Sutter, S.L. et al., Emergency Preparedness Source Term Development for the Office of Nuclear Material Safety and Safeguards-Licensed Facilities, NUREG/CR-3796, prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Washington, DC, 1984.

Sutter, S.L., "Aerosols Generated by Releases of Pressurized Powders and Solutions in Static Air," NUREG/CR-3093, 1983.

Tagg, B., "Determination of the Resuspension of Plutonium Arising from Contaminated Surfaces in a Pu Operational Area," in Proc. Into. Symp. Radiological Protection of the Worker by the Design and Control of His Environment, Bournemouth, Paper No. 26, 1966.

Tubis, M. and Wolf, W. Radiopharmacy, John Wiley and Sons, New York., 1980.

Valentine, A., Fitzgibbon, F., and Martinez, L., "Health Physics at the Los Alamos 'Wing 9' Hot-Cell Facility," in Health Physics Operational Monitoring, Vol. 1, C.A. Willis and J.S. Handloser, Editors, Gordon and Breach, NY, 1972, pp. 141-164.

Vincent, J.H. and Mark, D., "Comparison of Criteria for Defining Inspirable Aerosol and Development of Appropriate Samplers," Am. Indust. Hygiene Assn. Journ., 454-457, 1987.

Wald, N., Wechsler, R., Brodsky, A., and Yaniv, S., "Problems in Independent Medical Management of Plutonium-Americium Contaminated Patients," in Diagnosis and Treatment of Deposited Radionuclides, H. A. Kornberg and W. D. Norwood.

Wiatrowski, W.A., Cooke, E.P., Kopp, D.T., and Jordan, D.W., "Radiocontamination in Medical Centers from Diagnostic Nuclear Medical Procedures," Health Physics, Vol. 47, No. 2, pp. 297-298, 1984.

U.S. Pharmacopeia and the National Formulary, 1985.

APPENDIX B: DERIVATION OF EMISSION FACTORS FROM IFTAH DATA

Several of the emission factors presented in Section 3.2 are derived from measurements of the inhaled fraction of the activity handled (IFTAH). The IFTAH is defined as the ratio of the amount of activity inhaled by a worker to the amount of activity handled during the exposure. In order to derive an emission factor from an IFTAH, it is also necessary to know the fraction of the activity made airborne from the process which is inhaled by the worker (F). Once a value for F is obtained, the emission factor can be derived from:

$$\text{Emission Factor} = \text{IFTAH} / F .$$

Since measured values for F are not available, it is necessary to estimate them theoretically. Consider a worker handling material on an open bench in a room with no ventilation and with relatively "still" air moving at a speed of 15 feet per minute (HE64, SU84). Assume that a release occurs at a point on the bench that is at a distance r from the worker's breathing zone, and that an amount q of respirable material is released to the air. Further assume that, on the average, the ejected material is released uniformly in all directions into the upper hemisphere of radius r, and that the air stream that carries the material is sufficiently violent to propel the initial volume of material to distance r instantaneously. This sudden ejection of the material to distance r tends to maximize the quantity of material inhaled before natural convection dissipates the concentration. The re-entry of material into the breathing zone with time is neglected in this analysis, since even minimal ventilation will rapidly carry the airborne material into the ventilation system.

The concentration inside the hemispherical puff of $(2/3)\pi r^3$ at $t = 0$ is then be taken to be

$$C_0 = q/(2\pi r^3/3). \quad (1)$$

Assume that at distance r , the "boundary convection area" (the surface of equal concentration), the velocity vector has magnitude v , and half the time points away from the source, and half the time toward the source of material. (Convection currents move in random directions.) This allows us to calculate an effective dilution rate following the initial release of material, which is not likely to extend beyond about 1 meter before it is dissipated, regardless of the initial particle and air velocity (HE64).

The differential fractional dilution per unit time of the concentration within the boundary convection area then becomes the fractional amount of replacement air entering the boundary convection area through one-half the area, which is

$$\begin{aligned} - dC/Cdt &= 0.5 v 2\pi r^2/(2\pi r^3/3) \\ &= 3 v/2r \end{aligned} \quad (2)$$

By integrating Eq. 2 from $t = 0$ to a variable time t , and from C_0 to $C(t)$, we obtain the concentration as a function of time:

$$C(t) = q \exp(-\lambda t)/(2\pi r^3/3) \quad (3)$$

If we assume a uniform breathing rate, b cm³ air per minute, then the total amount I of activity inhaled is given by

$$I = \int_{t=0}^{t=\infty} b C(t) dt = (3bq/2\pi r^3) \int_{t=0}^{t=\infty} \exp(-3v/2r) t dt$$

$$= bq/\pi r^2 v \quad (4)$$

The fractional intake of the amount q released to the air in the puff is then given by

$$F = I/q = b/\pi r^2 v . \quad (5)$$

Letting $b = 20,000$ cm³/minute (20 breaths per minute times 1,000 cm³ of air per inspiration) and $v = 15$ feet/minute = 7.64 cm/sec, we obtain a conservative relationship for the value of F as a function of distance between the source and the breathing zone

$$F = 333/7.64\pi r^2$$

$$= 13.9/r^2. \quad (6)$$

For $r = 1$ meter (100 cm), $F = 0.0014$; for $r = 0.66$ meters (66 cm, about 2 feet), $F = 0.0032$; and for $r = 30.5$ cm (1 foot), $F = 0.015$. It is not likely that the puff will be maintained in significant concentrations beyond a distance of approximately one meter, and Hemeon assumes on the basis of industrial experience that "arm's length work" takes place at 2 feet and "close work" at 1 foot (HE64).

At the breathing rate assumed, inspirations and expirations each take approximately 1.5 seconds. Air moving at 7.6 cm/sec will move approximately 12 cm in 1.5 seconds. The radius of the 1-liter volume of air inhaled is

$$r = (3\pi v/4)^{1/3} = 6.2 \text{ cm.} \quad (7)$$

Further, the mean time for removal of a particle from the breathing zone consistent with this model is given by

$$\tau = 1/\lambda = 2r/3v = 8.7 \text{ sec at 100 cm,} \quad (8)$$

and the half-time for decrease of the concentration at 100 cm is

$$T_{1/2} = 0.693/\lambda = 6 \text{ sec.} \quad (9)$$

This is only two breathing cycles at 100 cm. Thus, if a person is exhaling rather than inhaling when the release occurs (and thus also diluting further the concentration by creating additional air movement), then the fractions inhaled will be smaller than the F values calculated above. On the other hand, if the worker is beginning a cycle of inhalation when the release occurs, then higher F values may be expected.

Considering all the factors in the above analysis, we estimate the following value for the upper limit of the fraction of the activity released to the air in respirable form that is inhaled by the worker

$$F_{\max} = 0.01. \quad (10)$$

The use of a maximum rather than a minimum value for F would appear contrary to our objective of deriving an upper-limit estimate for the emission factor. However, we are not interested in estimating the fractional release to air from a single observation; rather, our objective is to obtain an upper bound for a distribution of fractional releases. Thus, the following line of reasoning is applicable here.

The upper bound (e.g., the 95th percentile) of the underlying probability distribution of worker intakes would result from incidents that produce releases of large fractions of the material in process, combined with the large fractions inhaled of that amount released (which could, for example, result from processes involving proximity of the worker to the material combined with more frequent and vigorous agitation of the material). Thus, the "upper bound" fractional release of material to air (for the distribution of events) will be estimated using an "upper bound" estimate of fractional intake of material made airborne and the "upper bound" IFTAH values.

A mathematical statement may help to clarify the foregoing.

Let $p'(x)$ = the probability density function of fractional releases x , and

$p''(y/x)$ = the conditional probability density function of fractional intakes y of that released, given the fractional amount released is x .

In general, $p''(y/x)$ can be a function of x as well as y , since the more violent disturbances (such as explosions) that cause a release of activity to the air could also disturb the air currents that transport the material to a worker's breathing zone.

However, the influence of the value of x on air currents in the breathing zone can be assumed to be negligible for the smaller releases.

Now, the probability density of fractional intake i , $P(i)$, of an amount of radioactive material placed into process is given by

$$P(i) = p'(x) p''(y/x), \quad (11)$$

where

$$i = x y. \quad (12)$$

The probability, P' , of an intake fraction greater than or equal to $i = I$ is then given by

$$P'(i \geq I) = 1 - \int_{i=0}^{i=I} P(i) di \quad (13)$$

$$= 1 - \int_{i=0}^{i=I} P(xy) ((di/dx)dx + (di/dy)dy), \quad (14)$$

since x and y are assumed to be independent random variables. Thus, substituting new limits of the variables of integration, Eq. 13 becomes

$$P'(i \geq I) = 1 - \int_{x=0}^{x=I/y} P(xy) y dx - \int_{y=0}^{y=I/x} P(xy) x dy \quad (15)$$

It is noted that x , y , and $P(i)$ can never be negative. Therefore, higher intake values, i , are more probable when both y and x are maximized together, so that the ranges of integration of both terms are minimized, and thus P' is maximized.